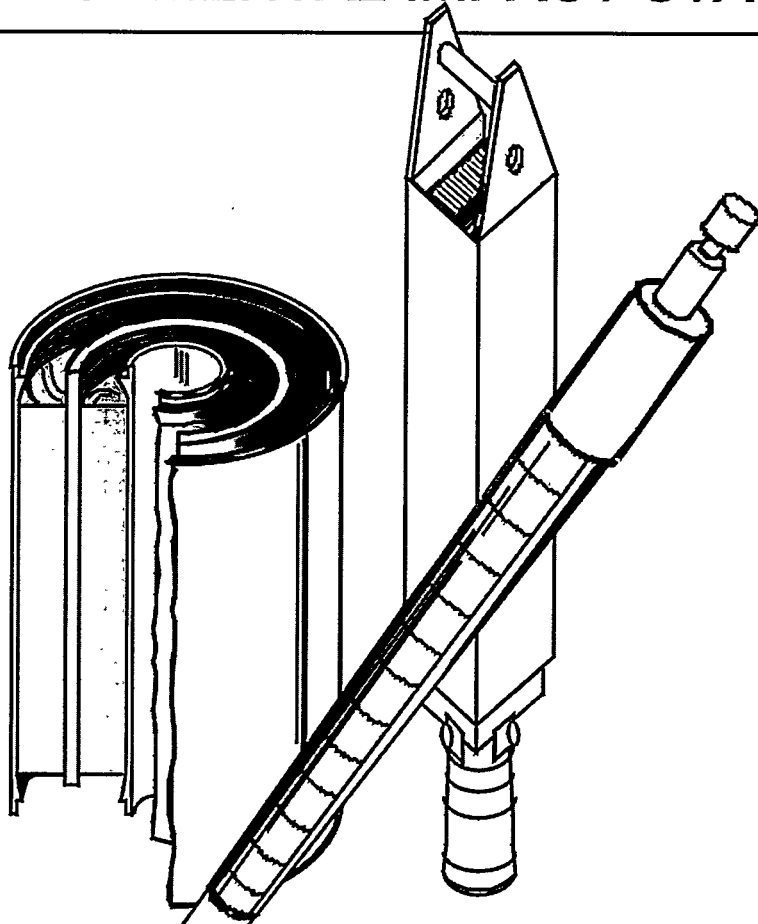


Savannah River Site

Spent Nuclear Fuel Management

FINAL ENVIRONMENTAL IMPACT STATEMENT



March 2000

Department of Energy • Savannah River Operations Office • Aiken, SC

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RESPONSIBLE AGENCY: U.S. Department of Energy (DOE)

TITLE: Savannah River Site, Spent Nuclear Fuel Management Final Environmental Impact Statement (DOE/EIS-0279)

EC

CONTACT: For additional information on this environmental impact statement, write or call:

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The EIS is also available on the internet at: <http://tis.eh.doe.gov/nepa/docs/docs.htm>.

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ABSTRACT: The proposed DOE action considered in this environmental impact statement (EIS) is to implement appropriate processes for the safe and efficient management of spent nuclear fuel and targets at the Savannah River Site (SRS) in Aiken County, South Carolina, including placing these materials in forms suitable for ultimate disposition. Options to treat, package, and store this material are discussed. The material included in this EIS consists of approximately 68 metric tons heavy metal (MTHM) of spent nuclear fuel (20 MTHM of aluminum-based spent nuclear fuel at SRS, as much as 28 MTHM of aluminum-clad spent nuclear fuel from foreign and domestic research reactors to be shipped to SRS through 2035, and 20 MTHM of stainless-steel or zirconium-clad spent nuclear fuel and some Americium/Curium Targets stored at SRS.

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Alternatives considered in this EIS encompass a range of new packaging, new processing, and conventional processing technologies, as well as the No Action Alternative. A preferred alternative is identified in which DOE would prepare about 97 percent by volume (about 60 percent by mass) of the aluminum-based fuel for disposition using a melt and dilute treatment process. The remaining 3 percent by volume (about 40 percent by mass) would be managed using chemical separation. Impacts are assessed primarily in the areas of water resources, air resources, public and worker health, waste management, socioeconomic, and cumulative impacts.

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PUBLIC INVOLVEMENT: DOE issued the Draft Spent Nuclear Fuel Management EIS on December 24, 1998, and held a formal public comment period on the EIS through February 8, 1999. In preparing the Final EIS, DOE considered comments received via mail, fax, electronic mail, and transcribed comments made at public hearings held in Columbia, S.C. on January 28, 1999, and North Augusta, S.C. on February 2, 1999. Completion of the Final EIS has been delayed because DOE has performed additional analyses of the melt and dilute technology, discussed in Chapter 2 and Appendix G. Comments received and DOE's responses to those comments are found in Appendix G of the EIS.

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Change Bars

Changes from the Draft EIS are indicated in this Final EIS by vertical change bars in the margin. The bars are marked TC for technical changes, EC for editorial changes, or if the change was made in response to a public comment, the designated comment number is as listed in Appendix G of the EIS.

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Abbreviations for Measurements

cfm	cubic feet per minute
cfs	cubic feet per second = 448.8 gallons per minute = 0.02832 cubic meter per second
cm	centimeter
gpm	gallons per minute
kg	kilogram
L	liter = 0.2642 gallon
lb	pound = 0.4536 kilogram
mg	milligram
μCi	microcurie
μg	microgram
pCi	picocurie
°C	degrees Celsius = $5/9$ (degrees Fahrenheit – 32)
°F	degrees Fahrenheit = $32 + 9/5$ (degrees Celsius)

Use of Scientific Notation

Very small and very large numbers are sometimes written using “scientific notation” or “E-notation” rather than as decimals or fractions. Both types of notation use exponents to indicate the power of 10 as a multiplier (i.e., 10^n , or the number 10 multiplied by itself “n” times; 10^{-n} , or the reciprocal of the number 10 multiplied by itself “n” times).

For example: $10^3 = 10 \times 10 \times 10 = 1,000$

$$10^{-3} = \frac{1}{10 \times 10 \times 10} = 0.001$$

In scientific notation, large numbers are written as a decimal between 1 and 10 multiplied by the appropriate power of 10:

4,900 is written $4.9 \times 10^3 = 4.9 \times 10 \times 10 \times 10 = 4.9 \times 1,000 = 4,900$

0.049 is written 4.9×10^{-2}

1,490,000 or 1.49 million is written 1.49×10^6

A positive exponent indicates a number larger than or equal to one; a negative exponent indicates a number less than one. | EC

In some cases, a slightly different notation (“E-notation”) is used, where “ $\times 10$ ” is replaced by “E” and the exponent is not superscripted. Using the above examples

$$4,900 = 4.9 \times 10^3 = 4.9\text{E}+03$$

$$0.049 = 4.9 \times 10^{-2} = 4.9\text{E}-02$$

$$1,490,000 = 1.49 \times 10^6 = 1.49\text{E}+06$$

Metric Conversion Chart

To convert into metric			To convert out of metric		
If you know	Multiply by	To get	If you know	Multiply by	To get
Length					
inches	2.54	centimeters	centimeters	0.3937	inches
feet	30.48	centimeters	centimeters	0.0328	feet
feet	0.3048	meters	meters	3.281	feet
yards	0.9144	meters	meters	1.0936	yards
miles	1.60934	kilometers	kilometers	0.6214	miles
Area					
sq. inches	6.4516	Sq. centimeters	sq. centimeters	0.155	sq. inches
sq. feet	0.092903	sq. meters	sq. meters	10.7639	sq. feet
sq. yards	0.8361	sq. meters	sq. meters	1.196	sq. yards
acres	0.0040469	sq. kilometers	sq. kilometers	247.1	acres
sq. miles	2.58999	sq. kilometers	sq. kilometers	0.3861	sq. miles
Volume					
fluid ounces	29.574	milliliters	milliliters	0.0338	fluid ounces
gallons	3.7854	liters	liters	0.26417	gallons
cubic feet	0.028317	cubic meters	cubic meters	35.315	cubic feet
cubic yards	0.76455	cubic meters	cubic meters	1.308	cubic yards
Weight					
ounces	28.3495	grams	grams	0.03527	ounces
pounds	0.4536	kilograms	kilograms	2.2046	pounds
short tons	0.90718	metric tons	metric tons	1.1023	short tons
Temperature					
Fahrenheit	Subtract 32 then multiply by 5/9ths	Celsius	Celsius	Multiply by 9/5ths, then add 32	Fahrenheit

Metric Prefixes

Prefix	Symbol	Multiplication Factor
exa-	E	1 000 000 000 000 000 = 10 ¹⁸
peta-	P	1 000 000 000 000 000 = 10 ¹⁵
tera-	T	1 000 000 000 000 = 10 ¹²
giga-	G	1 000 000 000 = 10 ⁹
mega-	M	1 000 000 = 10 ⁶
kilo-	k	1 000 = 10 ³
centi-	c	0.01 = 10 ⁻²
milli-	m	0.001 = 10 ⁻³
micro-	μ	0.000 001 = 10 ⁻⁶
nano-	n	0.000 000 001 = 10 ⁻⁹
pico-	p	0.000 000 000 001 = 10 ⁻¹²
femto-	f	0.000 000 000 000 001 = 10 ⁻¹⁵
atto-	a	0.000 000 000 000 000 001 = 10 ⁻¹⁸

SUMMARY

S.1 Introduction

The management of spent nuclear fuel (SNF) has been an integral part of the mission of the Savannah River Site (SRS) for more than 40 years. Until the early 1990s, SNF management consisted primarily of short-term onsite storage followed by processing in the SRS chemical separation facilities to produce strategic nuclear materials.

What is Spent Nuclear Fuel?

Spent nuclear fuel is fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated. When it is removed from a reactor, spent nuclear fuel contains some unused enriched uranium and radioactive fission products. Because of its radioactivity (primarily from gamma rays), it must be properly shielded. The fuel elements exist in many configurations. Generally, a fuel element is covered by a metal called cladding and is shaped into long rods, flat plates, or cylinders.

With the end of the Cold War, the U.S. Department of Energy (DOE) decided in April 1992 to phase out processing of SNF for the production of nuclear weapons materials. Therefore, the management strategy for this fuel has shifted from short-term storage and processing for the recovery of highly-enriched uranium and transuranic isotopes to stabilization, when necessary, and interim storage pending final disposition. Interim storage includes preparing SNF for disposal in any potential geologic repository.

In addition to the fuel already onsite, the SRS will receive SNF from foreign research reactors until 2009 and potentially could receive SNF from domestic research reactors until 2035. As a result, the safe and efficient management of SNF will continue to be an important SRS mission.

A key element in the decisionmaking process for SNF management is a thorough understanding of the environmental impacts that may result from the implementation of the proposed action. The National Environmental Policy Act of 1969 (NEPA), as amended, provides Federal decisionmakers with a process to use when considering potential environmental impacts of proposed actions.

National Environmental Policy Act of 1969:

An act that requires Federal agencies to consider in their decisionmaking process the potential environmental effects of proposed actions, and to analyze alternative approaches to meeting the need for agency action.

Environmental Impact Statement: A detailed environmental analysis of any proposed major Federal action that could significantly affect the quality of the human environment. It is a tool to assist the decisionmakers; it describes the positive and negative environmental effects of the proposed action and alternatives.

Alternatives: The range of reasonable alternative actions, that could be taken to meet the need for agency action.

Record of Decision: A concise public statement of the Federal agency's decision. It discusses the decision, identifies the alternatives considered, including the environmentally preferable alternative, and indicates whether all practicable means to avoid or minimize environmental harm were adopted (and if not, why not).

Following this process, DOE announced, on December 31, 1996 in the *Federal Register* its intent to prepare an EIS (61 FR 69085) and to establish a public comment period on the scope of the EIS that lasted until March 3, 1997. DOE accepted all comments received, even those received beyond the closing date. A public scoping meeting was held in North Augusta, South Carolina on January 30, 1997. Forty-one members of the public attended the meeting with 22 presenting comments or asking questions. In

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addition, during the scoping period DOE received letters, E-mails, and other written comments. Based upon these submittals and presentations, DOE identified 118 separate public comments which DOE divided into the following categories:

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- Processing of Spent Nuclear Fuel
- Alternative Technologies
- Need for a Transfer and Storage Facility
- Reuse of Nuclear Material for the Generation of Electricity
- Waste Form/Road-Ready Condition/Repository/Yucca Mountain
- Socioeconomic Impacts
- Human (Occupational and Public) Health
- Chemistry of Spent Nuclear Fuel
- Privatization
- Waste Generation
- No-Action Alternative
- Out-of-Scope Comments

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Utilizing input from the public scoping meeting and the NEPA process, DOE prepared a draft Environmental Impact Statement (EIS) for public comment.

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A Notice of Availability of the Draft EIS appeared in the Federal Register on December 24, 1998. Public meetings to discuss and receive comments on the Draft EIS were held on Thursday, January 28, 1999 at the Holiday Inn Coliseum, Columbia, SC and on Tuesday, February 2, 1999 at the North Augusta Community Center, North Augusta, SC. The public comment period ended on February 8, 1999. In the public meetings 17 individuals commented on the draft EIS. During the 45-day comment period DOE also received 15 letters commenting

on the Draft EIS. DOE also received seven letters commenting on the EIS after February 8, 1999, and the comments have been addressed in the final EIS.

For ease of discussing the comments in this Summary, DOE divided the comments into 12 major categories. The major points associated with the public comments and DOE's responses are summarized below.

Processing

Comments were received related to processing of SNF. These ranged from support of processing as a proven method for disposition of SNF to admonitions that the processing facilities (canyons) should be shut down immediately. A number of comments asked for clarification regarding the criteria used for determining when processing would be necessary for SNF currently in storage. Commentors also criticized the method by which DOE outlined the missions of the canyons, and several requested definite closure dates for the canyons.

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Response: The canyons at SRS cannot be shut down immediately because DOE is utilizing these facilities to stabilize nuclear materials. In this EIS, DOE proposes to use the canyons to process a relatively small amount (about 3 percent by volume or 40% by weight) of the SNF under consideration to eliminate the potential for certain health and safety problems. The basis for selecting the SNF proposed for processing is discussed in Section 2.4.3.2 of the Final EIS. DOE estimates the processing time would be less than 6 months in F Canyon and about 1 year in H Canyon. The proposed processing operations are within the current canyon schedule planning basis. In other words, the proposed SNF processing activities would not extend the planned canyon operations. However, establishing closure dates for the SRS canyons is beyond the scope of this EIS.

Alternatives

Comments were received regarding alternatives to conventional processing of SNF. The com-

ments ranged from support for alternatives to conventional processing to questions regarding the details of alternatives and their impacts. Commentors also questioned DOE's ability to develop a new technology to treat SNF in a timely manner.

Response: DOE evaluated a variety of technologies for managing aluminum-based SNF at the SRS. DOE considers that the range of technologies included in the EIS to be an appropriate reflection of the technologies available. DOE also considers the range of alternatives evaluated in the EIS to represent a reasonable set of the technology combinations that could have been evaluated. Public comments did not reveal an SNF management technology that DOE has not considered. The DOE has completed considerable research and development work for the proposed SNF alternative treatment technology (i.e., Melt and Dilute). DOE is committed to developing and demonstrating that technology for aluminum-based SNF as quickly as possible.

Waste Form

Comments were received relating to the acceptability in any geologic repository of the SNF waste form that would be produced using a new (alternative) SNF treatment technology. The principal concern was that waste acceptance criteria for a geologic repository have not been established. In this regard, the concern was that alternative technologies for the disposition of SNF may produce a product that will not meet the final repository criteria.

Response: Waste acceptance criteria describe the physical, chemical, and thermal characteristics to which SNF and associated canisters must conform for emplacement in a geologic repository. DOE has assessed the waste forms the primary new SNF treatment technologies (Melt and Dilute, and Direct Disposal/Direct Co-Disposal) would generate against potential repository preliminary waste acceptance criteria. DOE concluded that waste forms from both technologies would meet the preliminary crite-

ria. Section 2.2.1 of the Final EIS has been revised to discuss the issue in greater detail.

Impacts

Comments were received related to the calculation of impacts from the proposed actions. These comments ranged from expressions that specific impacts were negligible to comments that the impacts from past Site activities had been underestimated.

Response: The impact estimates in the Final EIS are based on data from Site operations or operating experience and the judgement of expert analysts. DOE believes that the analyses presented in the EIS are adequate for comparing SNF management alternatives.

Openness/Independent Review

Comments were received regarding independent reviews of the SNF treatment technologies and how they would be used in the decisionmaking process. The comments called for increased public involvement. Some comments also called for DOE to share technology development data, particularly regarding the requirements and performance of the off-gas system.

Response: DOE believes that the EIS process provides adequate opportunities for public involvement. For example, DOE has invited public comment and input for this process during scoping meetings and during the public comment period for the Draft EIS. Information regarding technology development that is referenced in the Final EIS is available at the DOE Public Reading Room, University of South Carolina at Aiken, Gregg-Graniteville Library, University Parkway, Aiken, South Carolina and at the DOE Freedom of Information Reading Room (Room 1E-190), Forrestal Building, 1000 Independence Ave., Washington, D.C. Additionally, information regarding the development of the new SNF treatment technologies, including the off-gas system that would be used to collect fission products that could evolve during operation of the Melt and Dilute process, may be requested from Randall Ponik, U.S. Depart-

Summary

ment of Energy, P.O. Box A, Aiken, South Carolina 29802, (803) 557-3263 or via E-mail at randall.ponik@SRS.gov. DOE has also solicited comments on the Melt and Dilute process from outside the Department. Contributing institutions include the University of South Carolina, the U.S. Nuclear Regulatory Commission (NRC), and the National Academy of Sciences. Their reports are publicly available, including in the DOE reading rooms, or upon request.

Cost

Comments were received regarding the potential costs of SNF treatment alternatives. These comments primarily questioned whether all costs and credits had been considered. This included the credits for the separation and sale of usable enriched uranium to the commercial nuclear power industry. Comments also were made regarding the adequacy of funding for the implementation of SNF treatment alternatives.

Response: DOE prepared a report on costs associated with aluminum-based SNF treatment technologies. DOE considered all appropriate factors to prepare the report. A discussion of uranium credits (i.e., cost recovery based on the sale of low-enriched uranium to the private sector) was included in the cost analysis. The results of the cost report are discussed in Section 2.6.5 of the EIS. DOE obtained an independent review of the cost report; the recommendations from the independent review were factored into the report.

A timeline has been established for the development, design and implementation of the melt and dilute facility. This timeline will be controlled through DOE's line item budget and procurement process pending completion of this NEPA process. The design and construction of a full-scale facility would need to be developed in the context of constrained, out-year budget targets, and funding for such a facility would need to be balanced against other priorities at SRS. DOE has developed a schedule that can be used as a baseline for near-term planning and budgeting purposes. The FY 2000 budget has been established and includes funding for design

completion of the L-Area Experimental Facility (LEF). The LEF is a pilot test facility which will demonstrate the feasibility of the melt and dilute technology. LEF is scheduled to be constructed and placed online by the end of FY 2002.

References

Comments were received related to the references used in the preparation of the EIS. The comments generally suggested alternate sources of information for the EIS and suggested that both foreign and domestic SNF handling experience be included in the discussion.

Response: DOE considered these suggestions. Based on the reports cited by the commentors, DOE believes accurate and current information was used to prepare the Final EIS. The information is based on actual Site operations (e.g., handling foreign and domestic SNF) and conditions or on estimates of operational activities and conditions that would exist for new facilities. As a result, DOE believes that data and references used in preparing the EIS provide an adequate basis for estimating impacts and for comparing technologies and alternatives. Current regulatory requirements and information have been cited as applicable.

Nonproliferation

Comments were received regarding nonproliferation issues as they relate to the treatment and disposition of SNF. A number of commentors felt that nonproliferation was being overemphasized in relation to its importance. However, one commentor doubted the independence of DOE in the preparation of the nonproliferation study, and another commentor stated that DOE should take a worldwide leadership role in nonproliferation by treating SNF without separating potential weapons materials.

Response: DOE believes nonproliferation to be one of the factors that must be considered during the decision-making process. DOE conducted a nonproliferation study for SNF treatment technologies in conjunction with the

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preparation of the Draft EIS. The study concluded that all technologies considered in the Draft EIS were compatible with the nonproliferation goals of the United States but that separations technologies, such as Conventional Processing, had distinct disadvantages because fissile material would be separated. The study was reviewed by experts independent of DOE: Matthew Bunn, Belfor Center for Science and International Affairs, Harvard University; Frank von Hippel, Woodrow Wilson School of Public and International Affairs, Princeton University; George Bunn, Center for International Security and Cooperation, Institute for International Studies, Stanford University; Harold Bengelsdorf, Bengelsdorf, McGoldrick and Associates, LLC; and David Albright, Institute for Science and International Security. No problems were identified with the conclusions presented in the report.

Methodology

Comments were received related to the methodologies used in the preparation of the EIS. These included both positive and negative comments on health issues and environmental justice. One of the commenters asked what environmental impact would result from the release of cesium into the atmosphere in the event that the filtration system does not capture all the cesium. Another commenter stated that DOE had minimized impacts in the Cumulative Impacts Chapter and only used a limited amount of available information regarding actual operating experience. The Environmental Protection Agency (EPA) commended DOE on its method of segregating spent fuel by type and then applying the appropriate treatment methodology as the best way to proceed.

Response: Impacts in the EIS are estimated from the best available information, including operational data whenever possible. When operations data do not exist, SRS relies on experience and information from similar facilities and the best judgement of technical experts.

Purpose and Need

Comments were received related to the stated purpose and need for agency action. The comments generally focused on long-term solutions to the problems SNF poses and noted that other nuclear materials that could be processed in SRS facilities should also be addressed.

Response: DOE proposes to manage SNF in such a way as to prepare aluminum-based SNF to meet the requirements for disposal in a geologic repository, and will make the SNF ready for offsite shipment. DOE is separately evaluating potential geologic disposal of high-level waste and spent nuclear fuel in the EIS for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada as discussed in Section 1.6 of the SNF Management EIS. DOE has addressed other nuclear materials that could be managed at the SRS as part of the cumulative impacts discussion in Chapter 5 of this EIS. The inventory of material was based on recent studies completed by DOE (see Chapter 5).

Safety

Comments were received related to general safety issues of the proposed actions. Most comments were related to concerns on whether or not facilities would be constructed and operated using stringent safety standards.

Response: DOE is committed to the protection of workers, the public, and the environment. All operations and facilities at SRS meet or exceed all applicable health protection and safety requirements. SNF treatment facilities and operations will meet or exceed all applicable requirements.

Failure of Stored SNF Before Treatment

Comments were received regarding the possibility that stored SNF could fail before proposed treatment facilities are available. The comments requested impact estimates for these potential failures.

Response: The preferred alternative in the Final EIS includes a discussion of the action that would be taken (processing in SRS canyons) should SNF fail while in storage pending implementation of a new treatment technology. Section 1.5 of the Final EIS includes a qualitative discussion of the types of health and safety issues (e.g., uncontrolled release of fission products into storage basin water) that would be created by the failure of the SNF that DOE believes presents certain vulnerabilities for continued storage.

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In addition, a number of other comments were received that offered editorial suggestions, could not be easily categorized, or were deemed to be out of scope of this EIS. Comments received and DOE's responses to all comments are presented in Appendix G of the EIS.

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After consideration of public comments, DOE prepared a Final EIS. Decisions on the management of SNF at SRS will be presented in a Record of Decision issued at least 30 days after the Notice of Availability of the Final EIS is published in the *Federal Register*. The Record of decision will be published in the *Federal Register*.

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S.2 Background

S.2.1 HISTORIC MISSIONS

The U.S. Atomic Energy Commission, a DOE predecessor agency, established the SRS in the early 1950s for the production of special radioactive isotopes to support national programs. Historically, the primary Site mission was the production of strategic isotopes (plutonium-239 and tritium) for use in the development and production of nuclear weapons. The SRS produced other isotopes (e.g., californium-252, plutonium-238, americium-241) to support research in nuclear medicine, space exploration, and commercial applications. DOE produced these isotopes in the five SRS production reactors. After the material was produced at the SRS, it was shipped to other DOE sites for fabrication into desired forms.

S.2.2 FUEL CYCLE

The material in the SRS reactors consisted of nuclear fuel and targets. The nuclear fuel was enriched uranium that was alloyed with aluminum and then clad with aluminum. The targets were either oxides or metallic forms of various isotopes such as neptunium-237 or uranium-238 that were clad with aluminum. Fuel and targets were fabricated at the SRS and placed in the reactors, and then the reactors operated to create the neutrons necessary to transmute the target material. After irradiation, the fuel and targets (collectively referred to as spent nuclear fuel) were removed from the reactors and placed in water-filled basins for short-term storage, about 12 to 18 months, before they were reprocessed in the SRS separations facilities.

During processing, SNF was chemically dissolved in F or H Canyon to recover the uranium or transuranic isotopes for future use. The remaining residue from the fuel, high-level radioactive waste consisting primarily of fission products and cladding in liquid form, was transferred to large steel tanks for storage. The high-level waste is currently being vitrified in the Defense Waste Processing Facility at the SRS to prepare it for placement in any potential geologic repository.

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S.2.3 CHANGING MISSIONS

In 1992, the Secretary of Energy directed that processing operations to produce strategic nuclear materials be phased out throughout the DOE complex. However, SNF and targets from previous production reactor irradiation cycles remained in storage at SRS spent nuclear fuel storage facilities.

In addition to nuclear material production missions, another mission for the SRS was (and continues to be) the receipt of SNF from DOE, domestic, and foreign research reactors. Historically, SNF from these reactors was stored in the Receiving Basin for Offsite Fuel at SRS. In the past, much of the research reactor SNF was reprocessed in the same manner as spent fuel from SRS production reactors. However, with

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the end of the Site's strategic nuclear materials production mission, SNF from research reactors has been accumulating in the Receiving Basin for Offsite Fuel and in the L-Reactor Disassembly Basin.

Some of the research reactor spent nuclear fuel sent to SRS was not aluminum based. Because DOE did not have the capability to reprocess that type of SNF at SRS, it was placed in wet storage at the Receiving Basin for Offsite Fuel, where it remains in storage.

By 1995 DOE was storing about 195 metric tons heavy metal (MTHM [metric tons heavy metal] – the mass of uranium in the fuel or targets, excluding cladding, alloy materials, and structural materials) – of aluminum-based SNF in the SRS reactor disassembly basins and the Receiving Basin for Offsite Fuel. DOE also was storing about 20 MTHM of non-aluminum-based SNF in the Receiving Basin for Offsite Fuel.

S.2.4 STABILIZATION

DOE has taken action to stabilize about 175 MTHM of the 195 MTHM of aluminum-based SNF that was in storage at SRS in 1995. DOE decided to stabilize this material following completion of the *Interim Management of Nuclear Materials Environmental Impact Statement* (DOE/EIS-0220). The primary purpose of the actions described in that EIS was to correct or eliminate potential health and safety vulnerabilities related to some of the methods used to store nuclear materials (including SNF) at SRS. In that EIS, DOE identified the remaining 20 MTHM (out of 195 MTHM) of aluminum-based SNF at SRS as "stable" (i.e., the SNF likely could be safely stored for about 10 more years, pending decisions on final disposition). That 20 MTHM of aluminum-based SNF is included in this EIS.

S.2.5 SPENT NUCLEAR FUEL CONSOLIDATION

In May 1995, DOE decided (60 FR 28680) under the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho*

National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement to consolidate existing and newly generated SNF at three existing Departmental sites (including SRS) based on fuel type, pending future decisions on ultimate disposition. DOE designated the SRS as the site that would manage aluminum-based SNF. As a result, DOE will transfer 20 MTHM of non-aluminum-based SNF from the SRS to the Idaho National Engineering and Environmental Laboratory (INEEL) and DOE will transfer about 5 MTHM of aluminum-based SNF at the INEEL to the SRS. Additionally, the SRS could receive about 5 MTHM of aluminum-based SNF from domestic research reactors through 2035.

In May 1996, DOE announced a decision (61 FR 25092) under the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (Nonproliferation Policy and Spent Fuel EIS) to accept about 18 MTHM of aluminum-based SNF containing uranium of United States origin from foreign research reactors for management in the United States at the SRS. The receipt of foreign research reactor SNF at SRS is now underway and receipts are scheduled to be completed by 2009. The 18 MTHM of foreign research reactor SNF that could be received at SRS is included in the scope of this EIS. (Recent decisions by some foreign research reactor operators have reduced the quantity of SNF expected to be shipped to SRS from about 18 MTHM to about 14 MTHM; however, the 18 MTHM projection is used for analysis purposes in this EIS because foreign research reactor operators still have the option to ship to the United States.) Table S-1 summarizes the amount of SNF to be managed at SRS that is considered in this EIS.

S.3 Purpose and Need for Action

DOE anticipates disposing of most of its aluminum-based SNF inventory in a geologic repository after treatment or repackaging. However, DOE does not expect a geologic repository to be

Summary

Table S-1. Quantity of SNF discussed in this EIS.

• Aluminum-based SNF stored at SRS	20 MTHM
• Domestic and DOE aluminum-based research reactor SNF to be received at SRS	10 MTHM
• Foreign Research Reactor aluminum-based SNF to be received at SRS	18 MTHM
• Non-aluminum-based SNF at SRS (to be shipped to INEEL)	20 MTHM

available until at least 2010 and shipments from DOE sites might not begin until about 2015. Until a repository is available, the Department needs to develop and implement a safe and efficient SNF management strategy that includes preparing aluminum-based SNF stored at SRS or expected to be shipped to SRS for disposition offsite. DOE is committed to avoiding indefinite storage at the SRS of this nuclear fuel in a form that is unsuitable for final disposition. Therefore, DOE needs to identify management technologies and facilities for storing and treating this SNF in preparation for final disposition.

S.4 Scope

In this EIS, DOE is evaluating the treatment and storage of about 48 MTHM of aluminum-based SNF including impacts from the construction and operation of facilities (either new or modified existing facilities) that would be used to receive, store, treat, and package SNF in preparation for ultimate disposition. Onsite transportation impacts are considered; however, no impacts associated with transporting SNF to SRS are included, because these impacts have been covered in other EISs. The potential impacts of transporting SNF to a geologic repository are discussed for completeness but no decisions related to transporting SNF offsite will be made under this EIS. Transportation of SNF to a federal repository will be addressed in the EIS for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye

County, Nevada (Notice of Availability of the Draft EIS published in 64 FR 44200, August 13 1999). The Yucca Mountain EIS is being prepared as part of the process to determine if the Yucca Mountain site is suitable as the site of the Nation's first geologic repository for SNF and high-level radioactive waste.

DOE also evaluates transferring 20 MTHM of non-aluminum-clad spent nuclear fuel currently stored in the Receiving Basin for Offsite Fuel at SRS to a new dry storage facility at SRS. This transfer would occur only if a dry storage facility were built as part of the implementation of a treatment technology to prepare aluminum-based spent nuclear fuel for disposition and if the dry storage facility became operational before the non-aluminum-clad fuel was transferred to the Idaho National Engineering and Environmental Laboratory. The transfer to dry storage would occur after the fuel had been relocated from the Receiving Basin for Offsite Fuel to the L-Reactor Disassembly Basin in support of activities necessary to phase out the use of the Receiving Basin for Offsite Fuel by fiscal year 2007.

This EIS does not evaluate the impacts of managing the non-aluminum-clad fuel at INEEL or of transporting the fuel to INEEL. These impacts were documented in the SNF management programmatic EIS (PEIS) and were evaluated as part of the process DOE used to decide to consolidate the storage of non aluminum-clad spent nuclear fuel at the INEEL.

SRS is storing Mark-51 and other targets in the Receiving Basin for Offsite Fuel (RBOF) in the Site's H-Area. This EIS evaluates the impacts of continuing to store the Mark-51 and other targets in RBOF, and evaluates an alternative of transferring them to dry storage to provide flexibility in material management operations.

DOE is evaluating potential uses for this material and the operations and facilities that would be necessary. The Mark-51 and other targets (described in Section 1.5 of this EIS) contain americium and curium isotopes that could be used to produce elements with higher atomic

TC

TC

TC

EC

TC

numbers such as californium-252. Californium-252 is used as a neutron source for radiography and in the treatment of certain types of cancer and for research in basic chemistry, nuclear physics, and solid-state chemistry. If DOE were to determine that a programmatic need for this material exists, the targets would continue to be stored at the SRS pending preparations to ship them to another DOE facility where isotope production capability currently exists or could be constructed. SRS does not have isotope production capability.

This EIS does not evaluate the impacts of utilizing target material for programmatic purposes such as production of californium. DOE would perform the appropriate National Environmental Policy Act review to evaluate the impacts of shipment of the targets to an isotope production facility and of construction (or modification) and operation of the production facility, should such a programmatic purpose be identified.

DOE is storing the Mark-18 targets in wet basins at the SRS. These targets are similar to the Mark-51 and other targets in that they contain americium and curium that could be used to produce elements with higher atomic numbers such as californium-252. They are different from the small (about two feet in length) Mark-51 and other targets because the Mark 18s are about 12 feet long and therefore have different requirements for storage, transportation and use. As is the case with the Mark-51 and other targets, DOE is not proposing any actions that would lead to programmatic use of the Mark-18 targets at this time. Because of their length, the Mark-18 targets would have to be reduced in size for use in production facilities at another DOE facility or transfer to

dry storage at the SRS. This EIS considers only continued wet storage of Mark-18 targets. However, the Interim Management of Nuclear Materials EIS (which is incorporated herein by reference) considered the alternative of processing the Mark-18 targets in the SRS canyons, should they present potential health and safety vulnerabilities. See Section 1.5 of this EIS for more information.

S.5 Decisions to be Based on this EIS

DOE expects to make the following decisions on the management and preparation of SNF for storage and ultimate disposition.

- The appropriate treatment or packaging technologies to prepare aluminum-based SNF that is to be managed at SRS. | EC
- Whether DOE should construct new facilities or use existing facilities to store and treat or package aluminum-based SNF that is expected to be managed at SRS. | EC
| TC
- Whether DOE should repackage and dry-store stainless-steel and zirconium-clad SNF pending shipment to the Idaho National Engineering and Environmental Laboratory, | EC
- Whether DOE should repackage and dry-store Mark-51s and other americium/curium targets in the event dry storage capability becomes available at SRS. | TC

S.6 Proposed Action

DOE's proposed action is to safely manage SNF that is currently located or expected to be received at SRS, including treating or packaging

TC

Summary

TC	aluminum-based fuel for possible offsite shipment and disposal in a geologic repository, and preparing non-aluminum-clad fuel and programmatic material (i.e., material that could be used in national programs) for dry storage or off-site shipment.	
EC	In the Record of Decision for the Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (61 FR 25092, May 17, 1996), DOE stated the Department would embark on an accelerated program at SRS to identify, develop, and demonstrate one or more non-processing, cost-effective treatment or packaging technologies to prepare aluminum-based foreign research reactor spent nuclear fuel for ultimate disposition.	EC
	Based on that decision, DOE's strategy is to select a new non-chemical processing technology or a new packaging technology that would put aluminum-based foreign research reactor SNF into a form or container suitable for direct placement in a geologic repository. Treatment or conditioning of the fuel would address potential repository acceptance criteria or safety concerns. Implementing the new non-chemical processing treatment or packaging technology would allow DOE to manage the SNF in a road-ready condition at SRS in dry storage pending shipment to a geologic repository.	EC
TC		
TC		
	Because of the similarity of the material, DOE proposes to manage the other aluminum-alloy SNF that is the subject of this EIS (domestic research reactor and DOE reactor fuels) in the same manner as the foreign research reactor fuels.	
EC	In the Record of Decision for the Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel, DOE stated that, should it become apparent by the year 2000 that DOE will not be ready to implement a new SNF treatment technology, DOE would consider chemically processing foreign research reactor SNF in F Canyon. DOE committed that any decision to use conventional chemical processing would consider the results of a study (62 FR 20001, December, 1998) on the nonproliferation, cost, and timing issues associated with chemically processing the fuel. DOE stated that any highly enriched uranium separated during chemical processing would be blended down to low enriched uranium.	EC
	With the limited proposed processing, as discussed below, and the current nuclear material stabilization program at SRS, DOE expects the canyons will be utilized to their fullest extent over the next several years. DOE has greater confidence now in the feasibility and availability of new non-chemical processing technologies than at the time the Nonproliferation Policy and Spent Nuclear Fuel EIS's Record of Decision was issued. Therefore, except in the case of potential health and safety vulnerabilities as discussed below, the use of the canyons for processing research reactor fuel as a backup for new technology would not be as likely.	EC
	DOE has included chemical processing as a management alternative in this EIS. However, DOE's strategy and preference is to use non-chemical separations processes. DOE proposes to use chemical separation processes when a potential health or safety vulnerability exists for aluminum-based SNF that DOE considers should be alleviated before a non-chemical separations process is in operation.	EC
	The limited proposed canyon SNF processing is not expected to extend the operating schedules for these facilities beyond the current planning basis. Processing would eliminate potential health and safety problems that could occur prior to the availability of a new SNF treatment technology. In the event the new treatment process becomes available, the SNF with potential health and safety vulnerabilities could be processed using the new treatment technology.	EC
	DOE may decide, in the future, that the Higher Actinide Targets have no programmatic use. Therefore, DOE proposes to maintain the Mark-18's, Mark-51's, and other Higher Actinide Targets pending decisions on final disposition.	TC

S.7 Categories of Spent Nuclear Fuel

DOE has categorized SNF at SRS into six groups (A through F), based on such characteristics as fuel size, physical or chemical properties, or radionuclide inventories. Table S-2 lists the amounts of each fuel type SRS expects to manage.

The aluminum-based fuels currently stored at SRS include some fuels that were not originally aluminum-clad (EBR-II and Sodium Breeder Experimental Reactor Fuel). Additionally, the aluminum-based category consists of one element not yet received but due to be shipped to SRS (the Advanced Reactivity Measurement

Facility Core Filter Block). Most of the fuels that were not originally aluminum-clad (but are included under this EIS's major category of aluminum-based fuel) have been declad and placed in aluminum cans. In their present form they can be processed at the SRS through the existing technologies on site. Other fuels at SRS which are non-aluminum-clad fuels cannot be processed in their existing form using the existing technologies and are categorized in this EIS as non-aluminum-based fuel. The Core Filter Block is included under the category of aluminum-based fuel since the most practical way of dealing with it (based on its unique configuration) is to process it utilizing the existing technology at SRS.

Table S-2. Spent nuclear fuel groups.

Fuel group	Volume (MTRE) ^a	Mass (MTHM) ^b
A. Uranium and Thorium Metal Fuels	610	19
B. Material Test Reactor-Like Fuels	30,800	20
C. HEU/LEU ^c Oxides and Silicides Requiring Resizing or Special Packaging	470 ^d	8
D. Loose Uranium Oxide in Cans	NA	0.7
E. Higher Actinide Targets	NA	<0.1
F. Non-Aluminum-Clad Fuels ^e	<u>1,900</u>	<u>20.4</u>
Total	33,780	68.2

- a. MTRE = Materials test reactor equivalent. An MTRE is a qualitative estimate of SNF volume that provides information on the amount of space needed for storage. An MTRE of Materials Test Reactor-Like Fuels would usually be one fuel assembly measuring about 3 inches by 3 inches by 2 feet long.
- b. MTHM = Metric tons of heavy metal.
- c. HEU = highly enriched uranium; LEU = low enriched uranium.
- d. Fuel group also includes about 2,800 pins, pin bundles, and pin assemblies.
- e. This fuel group will be shipped to Idaho National Engineering and Environmental Laboratory. It will not be treated at SRS.

Categorization of SNF at the Savannah River Site

- Group A: Uranium and Thorium Metal Fuels
Group B: Materials Test Reactor-Like Fuels
Group C: HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging
Group D: Loose Uranium Oxide in Cans
Group E: Higher Actinide Targets
Group F: Non-Aluminum-Clad Fuels

Uranium and Thorium Metal Fuels (Group A)

This group consists of fuels from the Experimental Breeder Reactor-II and the Sodium Reactor Experiment, as well as a core filter block from the Advanced Reactivity Measurement Facility at INEEL (that is scheduled to be transferred to SRS). This group also includes unirradiated Mark-42 targets that were manufactured from plutonium oxide-aluminum powder metal and formed into tubes that were clad with aluminum.

Summary

The Experimental Breeder Reactor-II fuel and Sodium Reactor Experiment fuel are uranium metal that has been declad and wet-stored in canisters in the Receiving Basin for Offsite Fuel at SRS. The declad fuel presents a potential health and safety vulnerability. These fuels have cores of reactive metals that were exposed when the fuel cladding was removed. Any contact of the reactive metal core with water would lead to relatively rapid oxidation of the core and disintegration of the fuel, resulting in the release of fission products and particulate fuel material to the water of the storage basin at SRS.

The unirradiated Mark-42 targets were manufactured from plutonium oxide-aluminum powder metal and formed into tubes that were clad with aluminum. The plutonium oxide and aluminum were pressed together in the manufacturing process. As a result, the unirradiated targets could be less durable than uranium-aluminum alloy SNF because of the particulate nature of the plutonium oxide, but more durable (i.e., less reactive) than uranium metal SNF since the plutonium is already in oxide form. The potential for dispersion of material into storage basin water in the event of cladding failure could present a health and safety vulnerability.

The core filter block at INEEL is made of depleted uranium and corrosion resistant metal (i.e., stainless steel), and was used as a neutron "filter" for reactivity experiments. As a result, the filter was subject to relatively short (or low-power level) exposure times in the test reactor and is only slightly irradiated.

EC | Material in this fuel group in its current form
TC | may not be acceptable for disposal in a repository due to the reactive nature of uranium metal or the particulate nature of some of the material.

This group accounts for approximately 2 percent of the volume of aluminum-based fuel that DOE is likely to manage at SRS from now until 2035. Because the fuel in Group A is made of unalloyed metal (i.e., it contains little or no aluminum), it is more dense than most of the other spent fuel considered in this EIS. As a result,

this small volume of fuel contains about 40 percent of the mass of heavy metal.

Materials Test Reactor-Like Fuels (Group B)

This group consists primarily of Materials Test Reactor fuels and other fuels of similar size and composition. Most research reactors – foreign and domestic – use Materials Test Reactor fuel. These fuels vary in uranium-235 content from just below 20 percent to about 93 percent. Approximately 70 percent of the Group B assemblies are highly enriched uranium (>20 percent uranium-235), and the remainder are low enriched uranium (<20 percent uranium-235). Group B accounts for approximately 97 percent of the volume of aluminum-based SNF that DOE will manage at SRS between now and 2035. DOE considers that there are no currently known health and safety vulnerabilities for this material that would preclude wet storage pending the operation of a new treatment technology.

Although some Group B fuels are stored at SRS in the Receiving Basin for Offsite Fuel or in L Reactor Disassembly Basin, at present most are at domestic universities, foreign research reactors, and DOE research facilities pending shipment to the Site.

HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging (Group C)

Fuels in this group are similar in composition to Group B fuels in that they are aluminum-based, highly enriched uranium and low enriched uranium oxides and silicides, but their size or shape might preclude packaging them in the disposal canisters proposed for use in a repository without resizing or special packaging considerations. Some fuel in this group is smaller in diameter and longer than Group B fuels or is larger than Group B fuels in both diameter and length; it often comes in odd shapes such as a 1.5-foot by 3-foot (0.46-meter by 0.9-meter) cylinder or a sphere with a diameter of 29 inches (74 centimeters). DOE would have to disassemble or use other volume-reduction activities to place such fuels in a nominal 17-inch direct co-disposal canister. At present, much of this fuel is at other

| TC

| EC

EC | DOE sites and in other countries but is scheduled to be received at SRS.

EC | A small amount of this fuel (currently stored in
TC | 14 cans) presents a potential health and safety vulnerability. The fuel was cut apart for research purposes and could release fission products and particulate material to the water of the wet storage basin at SRS should the storage cans leak. Additionally, fuel in this condition may not be acceptable in a geologic repository because the fuel is no longer intact.

EC | Together Group B and Group C fuels represent about 97 percent of the volume of all fuel to be treated at SRS.

Loose Uranium Oxide in Cans (Group D):

EC | This group consists of loose uranium oxide with
TC | fission products distributed throughout the material. The only material in this fuel group currently stored at the SRS is 676 cans of Sterling Forest Oxide. The majority of the material (estimated at over 6,000 cans) has not yet been produced at foreign research reactors. Research reactors in Canada would be the greatest single source for future material and these reactor operators are among those that, as discussed in Section S.2.4, may not participate in the foreign research reactor SNF return program. DOE expects that the material in this fuel group would not be acceptable for placement in a geologic repository because it is not in a tightly bound metal or ceramic matrix (i.e., it is a powder). Additionally, the Sterling Forest Oxide fuel presents a potential health and safety vulnerability due to the dispersible nature of the material should a storage can fail.

Higher Actinide Targets (Group E)

TC | This group contains irradiated and unirradiated target materials used to generate radionuclides with atomic numbers higher than that of uranium. The targets were aluminum-clad plutonium oxide that now contain significant quantities of americium and curium, which react under neutron irradiation to produce elements with still higher atomic numbers such as californium.

All materials in this group are stored in the Receiving Basin for Offsite Fuel. Group E accounts for less than 1 percent of the volume of aluminum-based SNF DOE will manage at SRS.

The Higher Actinide Target fuel group consists of 60 Mark-51 targets, 114 other targets, and 65 Mark-18 targets. This material was evaluated in the *Final Environmental Impact Statement for Interim Management of Nuclear Materials* (DOE/EIS-0220). Under the Record of Decision for the Final Environmental Impact Statement for Interim Management of Nuclear Materials (FR 65300, 12/19/95), DOE decided that the targets should remain in wet storage.

In this EIS, DOE evaluates the continued wet storage of the Mark-51 and other targets pending shipment offsite, or alternatively repackaging the Mark-51 and other targets to place them in a new dry storage facility so that the material could be transferred to dry storage if necessary to provide flexibility in spent fuel storage operations.

The Mark-18 targets are different from the Mark-51 and other targets in several ways. The most important distinction is that each Mark-18 target is one continuous piece about 12 feet long. The Mark-51 and other targets are about 2 feet long and could be handled, transported, and stored (including in a dry storage facility) in their current configuration. The 12-long Mark-18 targets would require size reduction for transportation or storage in a dry storage facility. The standard method to reduce the size of the Mark-18 targets would be to cut them up under water in an SRS wet storage basin. However, the condition of the Mark-18 targets presents a health and safety vulnerability for under water cutting because of the suspected brittle condition of the targets and the uncertainty of the region of the target assemblies that contains the target product (i.e., americium and curium) and fission products. The brittle condition is due to a very long irradiation cycle in a reactor at SRS. Cutting the targets using the existing Site capability could result in the uncontrolled release of radioactive material to the water of the Receiving Basin for Offsite Fuel. For these reasons, a

TC

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TC

Summary

EC | previous DOE assessment of this material (see
 EC | Section 1.6.2 of this EIS) concluded that the
 EC | Department should consider processing the
 TC | Mark-18 targets in F Canyon. These alterna-
 TC | tives are not included in this EIS because DOE
 TC | performed that evaluation in the *Final Environ-
 TC | mental Impact Statement for Interim Manage-
 TC | ment of Nuclear Materials*, incorporated herein
 TC | by reference. Those alternatives included dis-
 TC | solving the targets in F Canyon and then vitri-
 TC | fying the americium and curium in a new
 TC | F-Canyon vitrification facility, dissolving the
 TC | targets in F Canyon and recovering the ameri-
 TC | cium and curium as an oxide, and dissolving the
 TC | targets and transferring the americium and cu-
 TC | rium to the high-level waste tanks at the SRS.

Non-Aluminum-Clad Fuels (Group F):

This group consists of fuel that is clad in materials other than aluminum. It includes stainless-steel and zirconium-clad fuel at SRS that DOE plans to transport to the Idaho National Engineering and Environmental Laboratory in accordance with decisions based on the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE/EIS-0203).

S.8 Affected Environment

The SRS is in west-central South Carolina and occupies an area of approximately 300 square miles (approximately 800 square kilometers) adjacent to the Savannah River, primarily in Aiken and Barnwell Counties. The Site is approximately 25 miles (40 kilometers) southeast of Augusta, Georgia, and 20 miles (32 kilometers) south of Aiken, South Carolina. All alternatives described in this EIS, including the possible construction of new facilities to implement some of the alternatives, would occur within existing industrial areas at SRS.

S.9 Technologies**S.9.1 NEW SNF MANAGEMENT TECHNOLOGIES**

DOE has identified six reasonable new technologies to be analyzed in this EIS that could be used to prepare SNF at SRS for disposition. Most of the New Packaging Technology options and the New Processing Technology options are technologies that DOE has not previously applied to the management of aluminum-based SNF for the purpose of ultimate disposition. DOE assigned the highest confidence of success and greatest technical suitability to options that have relatively simple approaches.

Technology Options for Management of SNF at the Savannah River Site

- New Packaging Technology
 1. Direct Disposal/Direct Co-Disposal
 2. Repackage and Prepare to Ship to Other DOE Sites
- New Processing Technology
 1. Melt and Dilute
 2. Mechanical Dilution
 - Press and Dilute
 - Chop and Dilute
 3. Vitrification
 - Plasma Arc Treatment
 - Glass Material Oxidation and Dissolution System
 - Dissolve and Vitrify
 4. Electrometallurgical Treatment
- Conventional Processing Technology

S.9.1.1 New Packaging Technologies

Under the New Packaging Technology, two of the options, Direct Disposal and Direct Co-Disposal, are non-destructive methods to prepare and package aluminum-based SNF for placement in a geologic repository, while one technology option, Repackage and Prepare to Ship to Another DOE Site, is pertinent only to non-aluminum-clad SNF and higher actinide targets that are scheduled to be or could be shipped offsite.

EC

The Direct Disposal/Direct Co-Disposal process is relatively simple because the fuel would remain intact but be repackaged in a way that minimizes the possibility of criticality. Elaborate treatment processes and equipment would not be required. The dry storage method that would be used to store the fuel after repackaging is common for commercial SNF and is adaptable for aluminum-based SNF.

The Direct Disposal and Direct Co-Disposal technology options are discussed together in this EIS as Direct Disposal/Direct Co-Disposal. The only difference between the technologies is the diameter of the canister into which the SNF would be loaded. The Direct Disposal options would use a 24-inch diameter canister because this is the same size as the high-level waste canisters currently being produced at the SRS Defense Waste Processing Facility. The Direct Co-Disposal option would use a smaller diameter canister (17 inches) that could be placed in the void space at the center of high-level waste packages that will be assembled at the repository (i.e., a five-canister array of 24-inch diameter high-level waste canisters with one 17-inch diameter SNF canister placed in the center). In either case, the canisters would be stored at SRS and shipped from SRS in the same manner. In this EIS, Direct Disposal and Direct Co-Disposal are treated as the same technology and the final decision on canister diameter would be made during the engineering design phase of the project to implement the technology.

S.9.1.2 New Processing Technologies

DOE has identified four technology options that could treat aluminum-based SNF. These are: (1) Melt and Dilute, (2) Mechanical Dilution, (3) Vitrification, and (4) Electrometallurgical Treatment.

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The Melt and Dilute technology is more complicated than Direct Disposal since it would destroy the fuel elements, but it is one of the simplest of the destructive treatments. Under this technology, SNF would be melted along with other materials to ensure a low enriched

uranium-aluminum product. Most fission products would remain trapped within the product matrix, although some would be volatilized. The melt product would be sealed in corrosion-resistant canisters. DOE has substantial experience melting SNF on a small scale for research purposes and has not identified any reasons why a full-scale operation to melt aluminum-based SNF and dilute the highly-enriched uranium would not be achievable.

The Mechanical Dilution Technology would involve either the Press and Dilute or the Chop and Dilute options, which are similar. DOE has represented these two technologies for analysis as the Mechanical Dilution options.

In the Press and Dilute Technology, SNF would be crushed between layers of depleted uranium to produce a product with low overall enrichment. The product would be mixed with a neutron poison as necessary to prevent criticality. The final product would be sealed in special canisters.

In the Chop and Dilute Technology, SNF would be shredded and mixed with depleted uranium to produce a low enriched product. As in Press and Dilute, a neutron poison could be added as needed and the product sealed in special canisters.

Three SNF processing technologies, Plasma Arc Treatment, Glass Material Oxidation and Dissolution System, and Dissolve and Vitrify options all use processes that produce a product with properties similar to that produced at the Defense Waste Processing Facility at SRS. Therefore, DOE has represented these three as the Vitrification option.

In the Plasma Arc Treatment Technology, SNF would be melted by a high-temperature plasma torch in a furnace. The melted SNF would be mixed with a ceramic material to produce a glass-ceramic product. Depleted uranium would be included as necessary to reduce the enrichment of the final product, which would be sealed in special canisters.

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Summary

In the Glass Material Oxidation and Dissolution Technology, the SNF would be converted directly to borosilicate glass. Depleted uranium would be included as necessary to reduce the enrichment of the final product, which would be sealed in special canisters.

In the Dissolve and Vitrify Technology, SNF would be dissolved as in conventional processing, but the enriched uranium would not be extracted. Instead, the dissolved solution would be vitrified. Depleted uranium would be included as necessary to reduce the enrichment of the final product, which would be sealed in special canisters. DOE expects that the resulting waste form would be acceptable for disposal in a geologic repository.

DOE prepared the current waste acceptance criteria using information available to date. DOE considers the criteria to be conservative. As repository designs evolve and more information is available on waste form performance under relevant repository conditions, the acceptance criteria will change. DOE currently is characterizing conditions at the Yucca Mountain site in Nevada as a possible site for development of a geologic repository. If a decision were made to develop Yucca Mountain, DOE would submit a license application to the U. S. Nuclear Regulatory Commission (NRC). The acceptance criteria developed at that time would be the basis for waste acceptance specifications in the license application. These specifications likely would be available before the melt and dilute facility would be operational and before the canyons cease operating. Final waste specifications will not be available until after the NRC approves construction of a repository and authorizes a license for DOE to receive and store SNF and high-level radioactive waste, prior to the beginning of repository operations.

Electrometallurgical Treatment is an electrorefining process that would separate highly-enriched uranium from the aluminum and fission products in the SNF. In the Electrometallurgical Treatment Technology, the SNF would be melted into metal ingots. Processing of the ingots first would remove the aluminum from

the material. Further processing would remove the uranium from the material. The remaining material would be oxidized and dissolved in glass and then sealed in special canisters. This is a process that DOE has been evaluating for the management of certain non-aluminum-based SNF at other DOE sites.

S.9.1.3 Technical Considerations in Selecting a New Technology Option for SNF Processing

Part of DOE's proposed action is to prepare SNF to meet the requirements that the Department anticipates will apply to material to be disposed of in a geologic repository. Any technology that DOE implements must be able to provide a product that is compatible with such criteria. DOE must rely on reasonable assumptions about what the acceptance criteria would include when making decisions on SNF treatment technologies. DOE anticipates that eventually it will place its aluminum-based SNF inventory in a geologic repository after treatment or repackaging.

One of the technical risks in implementing any of the new SNF technology options is the uncertainty surrounding the acceptability of DOE SNF for placement in a geologic repository. While DOE has documented preliminary acceptance criteria in the Waste Acceptance System Requirements Document (Rev. 3, 1999), the acceptance criteria will become more detailed. Final acceptance criteria will not be available until after DOE were to receive authorization from NRC to receive and possess SNF and high-level waste, based on criteria that meet NRC requirements. DOE-SR is working closely with NRC (the Federal agency that would license the operation of a geologic repository) to ensure that the final product from the selected SNF treatment technology would be acceptable at a repository.

Recognizing that repository disposal is the ultimate endpoint for the melt and dilute waste form, DOE-SR signed in August 1997 a Memorandum of Understanding with NRC for their review and feedback on the research effort that

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EC DOE-SR is conducting. DOE-SR has provided the NRC with several technical reports on the results obtained from the research effort. Based upon their initial review, NRC stated in a June 1998 letter that "both the direct co-disposal and melt-dilute options would be acceptable concepts for the disposal of aluminum-based research reactor SNF in the repository." Additionally, as research efforts yield new findings, DOE is providing the information to the NRC for their feedback and review.

EC The EIS has been revised to discuss in greater detail the expected repository acceptance criteria and compare the treatment technology products to these criteria. This information is discussed in Section 2.2.1.

S.9.2 EXISTING SNF MANAGEMENT TECHNOLOGIES

TC The Conventional Processing technology is the only existing SNF treatment technology available at SRS.

TC With this technology, DOE would process SNF in F or H Canyon directly from wet storage. The process would chemically dissolve the fuel and separate fission products from the uranium by solvent extraction. Conventional Processing would apply to all SNF, except most of the targets in the Higher Actinide Targets fuel group (specifically the Mark-51 and "other" targets) and the non-aluminum-clad fuels. Non-aluminum-clad targets would be shipped to INEEL as a result of previous decisions by DOE.

TC The Record of Decision for the Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel described the possible use of F Canyon for SNF processing based on a preliminary concept to consolidate all processing operations in one canyon. Subsequent review has shown that consolidating highly enriched uranium spent fuel processing operations in F Canyon would not be practical due to criticality considerations and process capacity restrictions

associated with the pluto-nium-uranium extraction system used in F Canyon. Thus, in this EIS, H Canyon is referenced in regard to chemically separating highly enriched uranium spent fuel.

S.10 Alternatives

Alternatives Considered

- Minimum Impact Alternative
- Direct Disposal Alternative
- Preferred Alternative
- Maximum Impact Alternative
- No-Action Alternative

Because of the differences in the characteristics of the SNF and the capabilities of the technologies, no single technology could be applied to all the SNF. Table S-3 lists the technologies appropriate for each of the six fuel groups.

Because of the many possible combinations of technologies and fuel groups (more than 700), DOE has chosen to evaluate a limited number of configurations (as alternatives). The alternatives illustrate the range of impacts that could occur from any configuration the decisionmakers might select. Table S-4 and the following paragraphs describe the five alternatives considered in this EIS. See Section S.11 for a detailed description of the preferred alternative.

- *Minimum Impact Alternative:* This alternative combines the technologies appropriate for each fuel group that DOE believes would result in the lowest overall impact.
- DOE recognizes that this alternative might not result in the lowest impact for each impact category (e.g., worker health and public health could be lowest, but radioactive waste generation could be higher) and that there are other reasonable technology

Table S-3. Fuel groups and analyzed technology options.

Fuel group	New Packaging Technology			New Processing Technology			Conventional Processing Technology
	1. Prepare for Direct Co-disposal	2. Repackage and Prepare to Ship ^a	3. Melt and Dilute	4. Mechanical Dilution	5. Vitrification Technologies	6. Electro-metallurgical Treatment	7. Conventional Processing in Canyons
A. Uranium and Thorium Metal Fuels	Yes ^b	No	Yes ^c	No	Yes	Yes	Yes
B. Materials Test Reactor-Like Fuels	Yes	No	Yes	Yes	Yes	Yes	Yes
C. HEU/LEU ^d Oxides and Silicides Requiring Resizing or Special Packaging	Yes	No	Yes	Yes	Yes	Yes	Yes
D. Loose Uranium Oxide in Cans	No	No	Yes	No	Yes	Yes	Yes
E. Higher Actinide Targets	No	Yes ^e	No	No	No	No	No ^f
F. Non-Aluminum-Clad Fuels ^g	No	Yes	No	No	No	No	No

a. This alternative describes repackaging for storage at SRS pending shipment offsite.
b. "Yes" indicates that the technology could be applied to the fuel group. "No" indicates that the technology should not be applied to the fuel group (see Sections S.9.1.3 and Tables 2-1 and 2-2 of the EIS).
c. Except for the core filter block that may be incompatible with the melt and dilute process.
d. HEU = highly enriched uranium; LEU = low enriched uranium.
e. The Mark-18 targets from Fuel Group E are not acceptable for repackaging as proposed in this EIS. See footnote f.
f. This entry is with respect to the Proposed Action of this EIS. Conventional Processing with a follow-on treatment (e.g., vitrification, oxidation, or disposal) has been evaluated for the Mark-18 target material in the *Final Environmental Impact Statement for Interim Management of Nuclear Materials* (DOE/EIS-0220).
g. In light of a previous decision by DOE to transfer this material to INEEL, only packaging for dry storage needs to be considered further.

Table S-4. Alternatives analyzed in this EIS.

Fuel Group	No-Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative
A. Uranium and Thorium Metal Fuels	Continued Wet Storage	Prepare for Direct Co-Disposal	Conventional Processing	Conventional Processing	Conventional Processing
B. Materials Test Reactor-like Fuels	Continued Wet Storage	Prepare for Direct Co-Disposal	Prepare for Direct Co-Disposal	Melt and Dilute	Conventional Processing
C. HEU/LEU Oxide and Sili- cides Requiring Resizing or Special Packaging	Continued Wet Storage	Prepare for Direct Co-Disposal	Prepare for Direct Co-Disposal	Melt and Dilute ^a	Conventional Processing
D. Loose Uranium Oxide in Cans	Continued Wet Storage	Melt and Dilute	Melt and Dilute ^b	Melt and Dilute ^b	Conventional Processing
E. Higher Actinide Targets	Continued Wet Storage	Repackage and Prepare to Ship to Another DOE Site	Repackage and Prepare to Ship to Another DOE Site ^c	Continued Wet Storage	Repackage and Prepare to Ship to Another DOE Site ^c
F. Non-Aluminum-Clad Fuels	Continued Wet Storage	Repackage and Prepare to Ship to Another DOE Site	Repackage and Prepare to Ship to Another DOE Site	Repackage and Prepare to Ship to Another DOE Site	Repackage and Prepare to Ship to Another DOE Site

- a. Conventional processing would be the preferred technology for the failed or sectioned Oak Ridge Reactor fuel, High Flux Isotope Reactor fuel, Tower Shielding Reactor fuel, Heavy Water Components Test Reactor fuel, and a Mark-14 target (i.e., <1 percent of the material in this fuel group).
- b. Conventional processing is the preferred technology for the Sterling Forest Oxide fuel (i.e., about 10 percent of the material in this fuel group).
- c. Mark-18 target assemblies (approximately 1 kilogram heavy metal) would undergo conventional processing.

configurations that would result in similar minimal impacts. DOE expects that the impacts of this combination would be representative of the lower bound of impacts from the Proposed Action. This scenario would utilize the New Packaging and New Processing Technologies.

The Uranium and Thorium Metal Fuels would be treated using the Direct Disposal/Direct Co-Disposal technology with more complicated treatment (i.e., hot-vacuum drying). DOE recognizes that there is technical uncertainty regarding the acceptability of this material (treated this way) in a repository because of the potential reactivity of the material; however, Direct Disposal/Direct Co-Disposal was postulated to represent minimum impacts based on the assumption that the waste form would be acceptable for disposal in a geologic repository. Materials Test Reactor-like Fuels and HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging would receive minimum treatment (i.e., cold-vacuum drying and canning or resizing) using the Direct Disposal/Direct Co-Disposal technology before being placed in dry storage. The loose uranium oxide in cans would be treated using the Melt and Dilute Technology.

DOE would continue to wet-store the Mark-51 and other Higher Actinide Targets at the SRS. DOE would continue to wet-store the non-aluminum-clad spent nuclear fuel at SRS until the material is shipped to the Idaho National Engineering and Environmental Laboratory. In the event the non-aluminum-clad fuel has not been transferred offsite by the time a dry storage facility is in operation at the SRS (to support the Melt and Dilute Technology), DOE could repackage the fuel and transfer the material to dry storage. To maintain operational flexibility DOE could transfer the Mark-51 and other targets to dry storage. DOE would maintain the Mark-18 targets in wet storage pending disposition decisions due to potential health and safety concerns associated

with the actions that would be required to repackage the Mark-18 target assemblies.

- *Direct Disposal Alternative:* This alternative combines the New Packaging and New Processing Technologies with Conventional Processing Technology. Materials Test Reactor-like Fuels and HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging (except for the failed or sectioned fuel) would receive minimum treatment (i.e., cold-vacuum drying and canning) using the Direct Disposal/Direct Co-Disposal technology before being placed in dry storage.

All material in the Uranium and Thorium Metals Fuel group, the Sterling Forest Oxide fuel from the Loose Uranium Oxide in Cans group, and the failed or sectioned fuel from the HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging group would be treated with Conventional Processing because this material presents potential health and safety concerns and probably would not be suitable for placement in a geologic repository. Melt and Dilute would be applied to the majority of the material in the Loose Uranium Oxide in Cans fuel group because that material could be received after a melt and dilute facility was available.

DOE would manage the Higher Actinide Targets and the non-aluminum-clad SNF as described in the Maximum Impact Alternative.

DOE's Preferred Alternative: The alternative which DOE believes would best fulfill its statutory mission and responsibilities, giving consideration to economic, environmental, technical, and other factors.

Preferred Alternative: This alternative combines a New Packaging Technology option, a New Processing Technology option, and the Conventional Processing Technology option. Materials Test Reactor-like Fuels, most HEU/LEU Oxides and Silicides

Requiring Resizing or Special Repackaging, and most Loose Uranium Oxide in Cans would be stored and then treated using the Melt and Dilute technology option when that option became available. The Conventional Processing Technology option would be used for the Uranium and Thorium Metal Fuels, about 10 percent of the HEU/LEU Oxides and Silicides Requiring Resizing or Special Repackaging; and about 10 percent of the Loose Uranium Oxide in Cans because of the potential health and safety vulnerability of continuing wet storage of those fuels while awaiting the availability of Melt and Dilute technology.

TC DOE is not proposing any actions that would lead to the programmatic use of the Higher Actinide Targets. Therefore, DOE will maintain the Mark-18, Mark-51 and other targets in wet storage until decisions are made on final disposition.

TC EC • *Maximum Impact Alternative:* This alternative would provide Conventional Processing for all SNF except the Mark-51 and other" targets and the non-aluminum-clad fuels already selected for offsite shipment. This alternative provides the upper bound on range of impacts from potential configurations because the analyses presented are conservative in that they assume that the entire SNF inventory would be processed in the separations facilities, which would produce the greatest impacts of all the treatment options.

TC DOE would manage the Mark-51 and other Higher Actinide Targets and the non-aluminum-clad SNF as described in the Minimum Impact Alternative. DOE would process the Mark-18 Higher Actinide Targets in F Canyon followed by vitrification of the americium and curium in the new F-Canyon Vitrification Facility as analyzed in the *Final Environmental Impact Statement for Interim Management of Nuclear Materials*.

- *No-Action Alternative:* The implementing regulations of NEPA require the inclusion of a No-Action Alternative. Under this alternative, DOE would continue to store the SNF in the wet basins at SRS even though this would not meet the purpose and need for action. To maintain safe conditions, DOE would take necessary actions to ensure safe storage in the basins, such as consolidation of fuel and upgrades of systems to ensure good water quality. As determined by the Record of Decision for the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE/EIS-0203), DOE would transport the Non-aluminum Clad Fuels to the Idaho National Engineering and Environmental Laboratory.

EC

S.11 Preferred Alternative

DOE proposes to implement several technologies to manage spent nuclear fuel at SRS. These technologies are Melt and Dilute, Conventional Processing, and Repackage and Prepare to Ship. Each of these technologies would treat specific groups of spent nuclear fuel, as described below. The technology and fuel group combinations form DOE's Preferred Alternative in this EIS. Figure S-1 provides a flowchart for the preferred alternative.

Melt And Dilute

Melt and Dilute is the preferred treatment for 97 percent by volume (60 percent by mass) of the aluminum-based SNF at the Savannah River Site.

DOE has identified the Melt and Dilute process as the preferred method of treating most (about 97 percent by volume or about 32,000 MTRE) of the aluminum-based SNF considered in this EIS. DOE will continue to pursue a research and development program leading to a demonstration of the technology in FY 2001 using full-size irradiated research reactor spent nuclear

TC

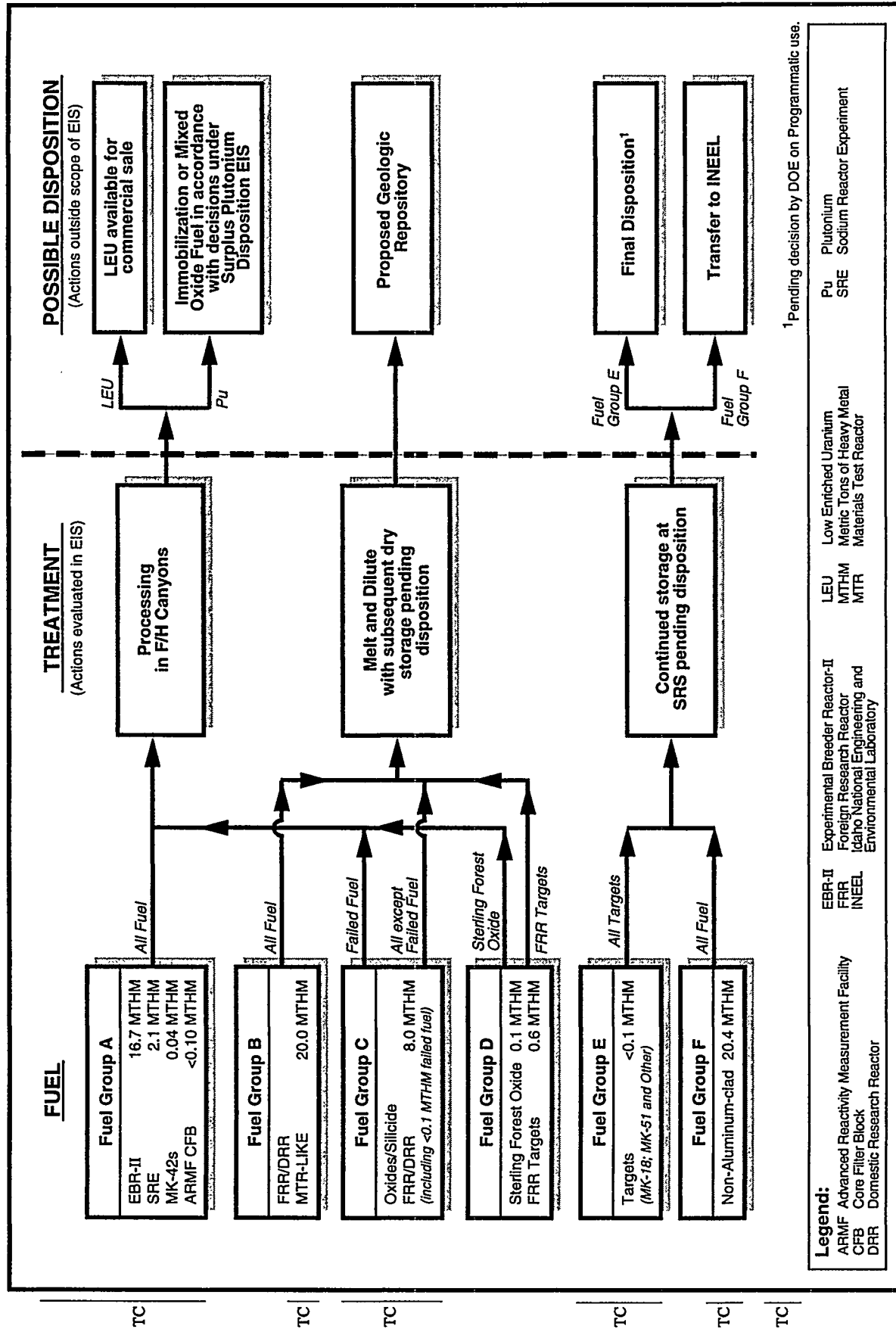


Figure S-1. Preferred Alternative Management Flow-Path.

	fuel assemblies. With a successful demonstration of the technology, DOE expects to have ready a treatment facility to perform production melt and dilute operations in FY 2008. DOE will ensure the continued availability of SRS conventional processing facilities until it has successfully demonstrated implementation of the Melt and Dilute treatment technology.	volume reduction is achieved because the melt and dilute process eliminates voids in the fuel elements and in the canisters and fuel baskets used in the Direct Disposal/Direct Co-Disposal technology. DOE considered Melt and Dilute to be among the most "proven" of the new non-separations-based technologies because DOE has extensive experience with fuel melting operations for research purposes.	EC
TC	The fuel proposed for the preferred Melt and Dilute technology includes the Material Test Reactor-like fuel, most of the Loose Uranium Oxide in Cans fuel, and most of the HEU/LEU Oxide and Silicide fuel. Exceptions are the uranium and thorium fuel, failed and sectioned oxide and silicide fuel, some loose uranium oxide in cans fuel, the Higher Actinide Targets, and non-aluminum-clad fuel.	The Melt and Dilute technology offers DOE the flexibility to engineer the final waste form to provide a high degree of confidence that the material would be acceptable for placement in a geologic repository. Major technical concerns such as fuel characterization, criticality control, and repository performance can be reduced or eliminated by tailoring the chemical and physical form of the final product to meet specific criteria. DOE expects the Melt and Dilute option would be relatively simple to implement and would be less expensive than other similar technology options, although the ongoing technology development initiative will determine the viability of this alternative. The major technical issue for implementing this technology would be the design of an off-gas ventilation system to capture volatilized fission products. Preliminary engineering studies indicate that the system could be designed using proven approaches for managing off-gases.	TC
TC	The Melt and Dilute Technology would satisfy DOE's objective and preference, as stated in the Record of Decision for the Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (60 FR 25091), to select a non-chemical separations-based technology to prepare aluminum-based SNF for placement in a geologic repository. Additionally, this new technology would provide significant waste reduction (of high-level, low-level, transuranic, etc.) in comparison to conventional chemical processing and is fully compatible with and supportive of the non-proliferation objectives of the United States. In the Melt and Dilute process, aluminum-based SNF would be melted and highly-enriched uranium would be diluted with depleted uranium to produce low-enriched uranium. No separation of fissile materials from fission products would occur.		TC
EC			
EC		To implement the preferred alternative (Melt and Dilute technology), DOE would construct a melt and dilute facility in the existing 105-L building at SRS and build a dry-storage facility in L Area, near the 105-L building. DOE is proposing to use this facility to house the Melt and Dilute process for the following reasons: the existing structure can accommodate the process equipment and systems; the applicable portions of the structure will meet DOE requirements for resistance to natural hazards (e.g., earthquakes); the integral disassembly basin has sufficient capacity for all expected SNF receipts and the current Site inventory; using 105-L avoids creating a new radiologically controlled facility that would eventually require decontamination and decommissioning; and	
EC	The potential impacts (e.g., worker and public health, waste generation, socioeconomics, etc.) among the new non-separations based technologies were all very similar; however, the Melt and Dilute option was the most efficient in volume reduction and produced the fewest number of SNF canisters. In fact, Melt and Dilute would volume reduce the fuel by more than 3 to 1 over Direct Disposal/Direct Co-Disposal. The		EC

Summary

EC | DOE has estimated the cost savings versus a new facility to be about \$70 million.

Using the Melt and Dilute technology, DOE would melt aluminum-based SNF and blend down any highly enriched uranium to low enriched uranium using depleted uranium that is currently stored at SRS. The material would be cast as ingots that would be loaded into stainless-steel canisters approximately 10 feet tall and 2 feet (or less) in diameter. The canisters would be placed in dry storage pending shipment to a geologic repository.

TC | During the development of the Melt and Dilute technology, DOE may determine that, for technical, regulatory, or cost reasons, the Melt and Dilute option is not viable. As a back-up to Melt and Dilute, DOE would continue to pursue the Direct Co-Disposal option of the New Packaging Technology and would implement this option if Melt and Dilute were no longer feasible or preferred. Direct Co-Disposal has the potential to be the least complicated of the new technologies and DOE believes this option could be implemented in the same timeframe as the Melt and Dilute option. However, DOE believes there is greater risk in attempting to demonstrate that aluminum-based SNF, packaged according to the Direct Co-Disposal option, would be acceptable in a geologic repository. A comparison of the preferred (Melt and Dilute) and back-up (Direct Co-Disposal) technologies DOE proposes to use to manage most of the aluminum-based SNF at SRS is presented in Table S-5.

EC | If DOE identifies any imminent health and safety concerns involving any aluminum-based SNF, DOE could use F and H canyons to stabilize the material of concern prior to the melt and dilute facility becoming operational.

Conventional Processing

Conventional Processing is the preferred treatment for 3 percent by volume (40 percent by mass) of aluminum-based SNF at the Savannah River Site.

DOE proposes to use conventional processing to stabilize some materials before a new treatment facility is in place. The rationale for this processing is to avoid the possibility of urgent future actions, including expensive recovery actions that would entail unnecessary radiation exposure to workers, and in one case, to manage a unique waste form (i.e., core filter block).

The total amount proposed for conventional processing is a relatively small volume of aluminum-based SNF at the SRS (about 3 % by volume and 40 % by mass). This material includes the Experimental Breeder Reactor-II fuel, the Sodium Reactor Experiment fuel, the Mark-42 targets and the core filter block from the Uranium and Thorium Metal fuel group; the failed or sectioned Tower Shielding Reactor, High Flux Isotope Reactor, Oak Ridge Reactor, and Heavy Water Components Test Reactor fuels and a Mark-14 target from the HEU/LEU Oxides and Silicides fuel group; and the Sterling Forest Oxide (and any other powdered/oxide fuel that may be received at SRS while H Canyon is still in operation) from the Loose Uranium Oxide in Cans fuel group. Although it is possible that a new treatment technology, such as melt and dilute, could be applied to most of these materials, DOE considers timely alleviation of the potential health and safety vulnerabilities to be the most prudent course of action because it would stabilize materials whose forms or types pose a heightened vulnerability to releasing fission products in the basin. Nonetheless, if these materials have not been stabilized before a new treatment technology becomes available, that new technology (melt and dilute) may be used rather than conventional processing.

The Experimental Breeder Reactor-II fuel and Sodium Reactor Experiment fuel are uranium metal that has been declad and stored in canisters in the Receiving Basin for Offsite Fuel. The declad fuels present a potential health and safety vulnerability. Should their existing storage containers leak, the metal fuel would corrode and release fission products to the water of the storage basin. Once the metal of the fuel is wetted, simply repackaging the fuel in a water-

Table S-5. Comparison of preferred and backup technologies for aluminum-SNF disposal.

Technology	Advantages	Disadvantages
Preferred technology: Melt-Dilute Process	• U-235 enrichment readily adjusted by dilution with depleted uranium to meet non-proliferation policy and nuclear criticality constraints. • Melting reduces the volume of the fuel (see Section A.2.1). DOE estimates about 400 canisters would be generated in comparison to about 1,400 canisters for Direct Co-Disposal. • Homogenous melt product provides basis for predictable behavior in a geologic repository.	• Implementation requires high temperature operation of melter and offgas control equipment in shielded cells.
Backup technology: Direct Co-Disposal Process	• Process technically straightforward to implement. Shielded-cell handling procedures well developed. • Meets non-proliferation policy criteria better than other technologies.	• Different SNF configurations, materials, and U-235 enrichments present packaging complexities. • No adjustment of U-235 enrichment possible to meet criticality constraints in a geologic repository. May require the use of exotic nuclear poisons. • No reduction in the volume of the fuel. • Non-uniform SNF structures and compositions complicates documentation of fuel characteristics to meet repository waste acceptance criteria and to predict behavior in a repository.

TC

TC

TC

TC

tight container would not arrest the corrosion and, in fact, could exacerbate storage concerns since potentially explosive hydrogen gas would continue to be generated inside the storage canister as the fuel continued to corrode. An instance of water intrusion and subsequent fuel corrosion has already occurred with one Experimental Breeder Reactor-II canister stored in the Receiving Basin for Offsite Fuel. Additionally, several problems have occurred with other uranium metal fuel in similar storage conditions at SRS (e.g., the Taiwan Research Reactor fuel with failed or missing cladding that was overpacked in canisters and stored in SRS wet basins). DOE addressed these situations by processing the failed or declad fuel in F Canyon to eliminate the health and safety vulnerability.

The failed or sectioned Tower Shielding Reactor, High Flux Isotope Reactor, Oak Ridge Reactor, and Heavy Water Components Test Reactor fuel, and a sectioned Mark-14 target from the HEU/LEU Oxides and Silicides fuel group also present potential health and safety vulnerabilities. The integrity of these fuels was destroyed for research purposes. Then the material was canned and placed in wet storage at SRS. A breach of or leak in the cans would expose the interior surfaces of the sectioned fuel to water, contaminating the water in the storage basin with radioactivity, and accelerating the corrosion of the fuel.

A potential health and safety vulnerability also exists for the unirradiated Mark-42 targets from the Uranium and Thorium Metal fuel group and

TC

the Sterling Forest Oxide fuel from the Loose Uranium Oxide in Cans fuel group. Should a breach occur in the cladding on the Mark-42 targets or in the canisters of Sterling Forest Oxide fuel, the particulate nature of the nuclear material in the targets and the Sterling Forest Oxide fuel could lead to dispersion of radioactive material in the water of the Receiving Basin for Offsite Fuel. Therefore, DOE is proposing to take action now to avoid the possibility of urgent future actions, including expensive recovery actions that also would entail unnecessary radiation exposure to workers.

DOE proposes to process the Experimental Breeder Reactor-II fuel and the Mark-42 targets in F Canyon. That fuel contains plutonium, approximately 114 kg of which would be recovered as part of the normal F Canyon chemical separations process and then transferred to FB-Line for conversion to metal. The plutonium metal would be considered surplus to the nation's nuclear weapons program and would be placed in storage at the SRS pending disposition pursuant to the January 2000 Record of Decision (ROD) for the Surplus Plutonium Disposition Environmental Impact Statement (DOE 1999). The surplus plutonium would be immobilized using the can-in-canister process or fabricated into mixed-oxide (MOX) commercial power reactor fuel at the SRS. DOE has scheduled processing of the Experimental Breeder Reactor-II fuel and the Mark-42 targets in FY00.

DOE proposes to process the Sodium Reactor Experiment fuel, the failed or sectioned fuel from the HEU/LEU Oxides and Silicides fuel group, and the Sterling Forest Oxide fuel in H-Canyon where the highly enriched uranium would be blended down to low enriched uranium and stored pending potential sale as feedstock for commercial nuclear fuel. DOE would begin processing operations in H Canyon in 2000 and could complete them in about 18 months.

DOE also proposes to process the core filter block from the Uranium and Thorium Metals fuel group. The core filter block is made of depleted uranium but it contains corrosion-

resistant metal (e.g., stainless-steel) that would be incompatible with the Melt and Dilute Technology for aluminum-based SNF. The core filter block could be processed in either F Canyon or H Canyon. In either case, the material would become feedstock to blend down highly enriched uranium from either conventional processing or melt and dilute operations.

The processing operations described above in both F and H Canyons would occur when the canyons were being operated to stabilize other nuclear material. It is the preference of the Department of Energy not to utilize conventional reprocessing for reasons other than safety and health. However, the core filter block is not compatible with the melt and dilute process for aluminum-based SNF. The benefit to develop a new process to accommodate this form(?) would be disproportionately small when compared to the cost (DOE 1998a). Consequently, the Department proposes an exception in this case.

Repackaging

Repackaging and dry storage is the preferred alternative for non-aluminum-clad SNF (about 6 percent by volume and 30 percent by mass of all the fuel considered in this EIS). Mark-51 targets, and other targets would be managed using onsite storage pending disposition decisions.

DOE would continue to wet-store the non-aluminum clad spent nuclear fuel at SRS until the material is shipped to the Idaho National Engineering and Environmental Laboratory. DOE could transfer the non-aluminum clad fuel to dry storage after the material had been relocated from the Receiving Basin for Offsite Fuel to the L-Reactor Disassembly Basin in support of activities to phase out operations in the Receiving Basin for Offsite Fuel by fiscal year 2007.

Continued Wet Storage

DOE is not proposing any actions that would lead to programmatic use of the Higher Actinide Targets. Therefore, the Mark-18, Mark-51 and

TC

TC

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other Higher Actinide Targets will be maintained in wet storage until decisions are made on their final disposition.

S.12 Comparisons of Environmental Impacts Among the Alternatives

Operational Impacts

Impacts from operations under all of the alternatives would have no effect on ecological resources, water resources, or cultural resources. The impacts from onsite transportation of SNF would be small under all alternatives.

Processing the Mark-18 targets (about 1 kilogram of heavy metal) was previously analyzed in the *Final Environmental Impact Statement on Interim Management of Nuclear Materials* and, therefore, was not analyzed in this EIS. The impacts of processing this small amount of material are minor and would not significantly add to the impacts currently analyzed for the Maximum Impact Alternative in this EIS. For example, total radiological dose from the Preferred Alternative to the maximally exposed individual for the entire period of analysis would be 0.67 millirem. Processing the Mark-18 targets would result in a dose of 0.0035 millirem. These extremely small doses are unlikely to result in any health effects.

Tables S-6 and S-7 list impacts for the five alternatives. The EIS identifies the following operational impacts with potential to be discriminators among the alternatives:

Worker and public health impacts – Estimated impacts are reported as latent cancer fatalities for the involved worker population, noninvolved worker, the maximally exposed member of the public, and offsite population (Table S-6). These impacts are summed over the period of analysis based on annual emissions and radiation doses.

Involved worker doses are estimated under the assumption that no worker would receive more than the SRS administrative annual limit of 500 millirem from normal

EIS Operational Impact Potential Discriminators

- Worker and Public Health Impacts
- Nonradiological Air Impacts
- Waste Generation
- Utilities and Energy Consumption
- Accidents

operations. The estimated latent cancer fatalities for the involved worker population for the entire period of analysis would be 0.28 for the Minimum Impact Alternative and 0.84 for the Maximum Impact Alternative.

The noninvolved worker highest estimated probability of a latent cancer fatality over the entire period of analysis would be 2.0×10^{-9} for the Minimum Impact Alternative and 6.3×10^{-7} for the Maximum Impact Alternative.

Table S-6 provides the incremental impact for health effects to the noninvolved worker, maximally exposed individual, and the off-site population above the current baseline for the operations of the wet storage basins at the SRS (the No-Action Alternative) over the entire period of analysis. Summing these baseline and incremental values is conservative because there would not be two SNF wet basins operating over the entire 38-year period of analysis.

The estimated latent cancer fatality probability to the maximally exposed individual over the entire period of analysis would be 3.0×10^{-10} for the Minimum Impact Alternative and 3.4×10^{-7} for the Maximum Impact Alternative. The

Table S-6. Impact summary by combination strategy.

	Parameter	No Action Alter- native (baseline)	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alter- native	Maximum Impact Alternative
Health Effects for Entire Period of Analysis (1998-2035)						
TC	Integrated latent cancer fatality probability for the noninvolved worker	$1.7 \times 10^{-6(a)}$	2.0×10^{-9}	9.6×10^{-9}	6.1×10^{-7}	6.3×10^{-7}
TC	Integrated latent cancer fatality probability for the maximally exposed member of the public	$3.1 \times 10^{-7(a)}$	3.0×10^{-10}	3.6×10^{-9}	9.5×10^{-8}	3.4×10^{-7}
	Integrated latent cancer fatalities for the worker population	0.30	0.28	0.34	0.33	0.84
	Integrated latent cancer fatalities for the general public	$1.1 \times 10^{-2(a)}$	1.1×10^{-5}	3.8×10^{-5}	3.4×10^{-3}	4.4×10^{-3}
Waste Generation for the Entire Period of Analysis (1998-2035)						
TC	Liquid (cubic meters)	2,300	660	1,200	1,050	10,500
	High-level waste generated (equivalent DWPFb canisters)	38	11	20	17	160
	Transuranic waste generated (cubic meters)	0	15	360	563	3,700
	Hazardous and mixed low-level waste gen- erated	76	25	46	103	267
	Low-level waste generated (cubic meters)	57,000	20,000	31,000	35,260	140,000
Utilities and Energy for the entire period of analysis (1998-2035)						
	Water consumption (millions of liters)	1,100	660	1,400	1,186	8,000
	Electricity consumption (megawatt-hours)	46,000	27,000	81,000	116,000	600,000
	Steam consumption (millions of kilograms)	340	190	520	650	3,600
	Diesel fuel consumption (thousands of liters)	230	180	2,300	2,760	22,000
EC	SNF Disposal Canisters (1998-2035)	0	~1,400	~1,300	400	0 ^c

a. Reflects current reactor-area emissions (including two SNF wet basins) for the entire period of analysis.

b. DWPF = Defense Waste Processing Facility.

c. The technology used in the Maximum Impact Alternative (i.e., Conventional Processing) would not produce any canisters of SNF.

Table S-7. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for each alternative (percent of regulatory standard).

No Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative
0.03 (nitrogen oxides)	0.07 (ozone [as VOC])	1.2 (nitrogen oxides)	1.1 (nitrogen oxides)	3.6 (nitrogen oxides)

VOC = volatile organic compound.

estimated offsite latent cancer fatalities would be 1.1×10^{-5} for the Minimum Impact Alternative and 4.4×10^{-3} for the Maximum Impact Alternative. The estimated latent cancer fatalities in the offsite population affected by SRS over the entire period of analysis would be much less than 1 for any alternative.

- *Nonradiological Air Quality* – Table S-7 presents the estimated maximum incremental concentration of the nonradiological air pollutant that would contribute the most to the deterioration of air quality at the SRS boundary for each alternative. As noted from Table S-7, the concentration of the nonradiological constituent contributing the highest fraction of the offsite air quality standard would range from 0.03 percent of the standard for the No-Action Alternative to 3.6 percent of the standard for the Maximum Impact Alternative. Under all alternatives, nonradiological air concentrations at the SRS boundary would be well below applicable standards.
- *Waste generation* – Wastes volumes were estimated over the period of analysis. The Maximum Impact Alternative would generate the greatest volume of waste, while the Minimum Impact Alternative would generate the least volume of waste (Table S-6). For wastes generated under all alternatives, DOE would use the surplus capacity in existing SRS waste management facilities to treat, store, dispose, or recycle the waste in accordance with applicable regulations.
- *Utilities and energy consumption* – The quantities of water, electricity, steam, and

diesel fuel that would be required over the entire period of analysis were estimated (Table S-6).

The Maximum Impact Alternative would require the most water, electricity, steam, and diesel fuel, while the Minimum Impact Alternative would require the least. For all alternatives, water and steam would be obtained from existing onsite sources and electricity and diesel fuel would be purchased from commercial sources. These commodities are readily available and the amounts required would not have an appreciable impact on available supplies or capacities.

- *Accidents* – DOE evaluated the impacts of potential accidents related to each of the alternatives. For each potential accident, the impacts were evaluated as radiation dose to the noninvolved worker, radiation dose to the offsite maximally exposed individual, collective radiation dose to the offsite population, and latent cancer fatalities to the offsite population. Table S-8 presents the results of this analysis. Table S-8 also indicates the estimated frequency of occurrence for each accident.

The highest consequence accident postulated under the continued wet storage, direct co-disposal, and repackaging and prepare to ship technologies is a seismic/high wind-induced criticality, which is estimated to result in 6.2 latent cancer fatalities in the offsite population. The highest consequence accident under conventional processing technology is a coil and tube failure with an estimated offsite population impact of 39

TC

latent cancer fatalities. The frequencies of these accidents are once in 2,000 to once in 26,000 years.

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For the other new SNF technologies evaluated, the maximum consequence accident (earthquake induced spill with loss of ventilation) is associated with the melt and dilute process. This accident is estimated to occur once in 200,000 years and to result in 10 latent cancer fatalities in the offsite population.

Construction Impacts

Impacts of construction would be minor and short-lived.

Construction activities could affect four resources: surface water, air, ecological resources, and socioeconomics. However, because workers would build the facilities needed to carry out the proposed action in an area of the Site that is already industrialized, DOE expects little impact to these resources from construction activities.

In summary, none of the alternatives analyzed would result in undue adverse environmental effects. The preferred alternative is the alternative that DOE considers provides the greatest assurance of preparing the SNF for ultimate placement in a geologic repository by using a relatively simple new processing technology and a proven technology.

S.13 Cumulative Impacts

L3-8

DOE evaluated the cumulative impacts of SNF management activities coupled with other past, present, and reasonably foreseeable future actions that could impact the SRS and its environs.

This cumulative impacts analysis included the impacts from SNF management, other related

DOE NEPA actions, current SRS operations, and potential processing in the SRS canyons of other nuclear materials located at other DOE sites. DOE analyzed cumulative impacts for the following areas: (1) air resources, (2) water resources, (3) public and worker health, (4) waste generation, and (5) utilities and energy consumption. Table S-9 presents the results of the non-radiological air resources cumulative impact analysis. Table S-10 presents the results of the cumulative analysis for the other technical discipline areas.

L3-8

S.14 Other Factors

DOE evaluated other factors such as technical availability, nonproliferation and safeguards, labor availability and core competency, custodial care, and cost. These factors are discussed in Section 2.6 of the Final EIS.

Life-cycle costs (1998 billion of dollars) for each of the alternatives were estimated as follows:

- | | |
|-------------------------------|-----|
| • Minimum Impact Alternative | 1.9 |
| • Direct Disposal Alternative | 1.9 |
| • Preferred Alternative | 2.0 |
| • Maximum Impact Alternative | 2.0 |
| • No Action Alternative | 1.7 |

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Life-cycle cost comparisons indicate that the No Action Alternative would be the least expensive. However, the cost of continued wet storage does not include costs of actions necessary to prepare SNF for ultimate disposition. The Direct Disposal Alternative and the Preferred Alternative (both using a renovated reactor building) have approximately the same life-cycle cost. Installation in a renovated reactor facility presents cost advantages of about \$70 million compared to a new treatment facility.

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Table S-8. Estimated maximum consequence accident for each technology.

Option	Accident Frequency	Consequences			
		Noninvolved Worker (rem)	MEI (rem)	Offsite Population (person-rem)	Latent Cancer Fatalities
Continued Wet Storage (No Action) ^a					
RBOF (high wind-induced criticality)	Once in 26,000 years	13	0.22	12,000	6.2
L-Reactor basin (basin-water draindown)	Once in 500 years	0.014	0.016	(b)	(b)
Direct Co-Disposal					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Repackage and Prepare to Ship					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Conventional Processing					
Processing phase in F/H Canyons (coil and tube failure)	Once in 14,000 years	13	1.3	78,000	39
Melt and Dilute					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Melt and dilute phase (earthquake-induced spill)	Once in 200,000 years	30	0.5	21,000	10
Mechanical Dilution					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Mechanical dilution phase (criticality with loss of ventilation)	Once in 33,000 years	0.71	0.074	3,000	1.5
Vitrification Technologies					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Vitrification phase (earthquake-induced release with loss of ventilation)	Once in 200,000 years	0.10	0.0017	71	0.035
Electrometallurgical Treatment					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Electrometallurgical phase (earthquake induced spill with loss of ventilation)	Once in 200,000 years	30	0.5	21,000	10

MEI = Maximally Exposed Individual.

RBOF = Receiving Basin for Offsite Fuels.

a. All alternatives would use RBOF and the L-Reactor Disassembly Basin; therefore, accidents in these facilities are possible for each technology.

b. Not available.

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Table S-9. Estimated maximum cumulative ground-level concentrations of nonradiological pollutants (micrograms per cubic meter) at SRS boundary.

Pollutant	Averaging time	SCDHEC ambient standard ($\mu\text{g}/\text{m}^3$)	Cumulative concentration ($\mu\text{g}/\text{m}^3$)	Percent of standard
Carbon monoxide	1 hour	40,000	10,093	25
	8 hours	10,000	6,921	69
Oxides of Nitrogen	Annual	100	33.1	33
Sulfur dioxide	3 hours	1,300	1,206	93
	24 hours	365	351.7	96
	Annual	80	34.1	43
Ozone	1 hour	235	1.8	1
Lead	Max. quarter	1.5	0.03	2
Particulate matter (≤ 10 microns aerodynamic diameter)	24 hours	150	130.4	87
	Annual	50	25.1	50
Total suspended particulates ($\mu\text{g}/\text{m}^3$)	Annual	75	67.1	89

Table S-10. Cumulative impacts.

Parameter	Cumulative total
Radiological Air Impacts	
Annual MEI ^a Dose (rem)	1.0×10^{-4}
MEI LCF ^b Probability (unitless)	5.1×10^{-8}
Annual Population dose (person-rem)	5.6
Population LCFs (unitless)	2.8×10^{-3}
Radiological Water Impacts	
Annual MEI Dose (rem)	2.4×10^{-4}
MEI LCF Probability (unitless)	1.2×10^{-7}
Population dose (person-rem)	2.6
Population LCFs (unitless)	1.3×10^{-3}
Worker and Public Health (Air and Water)	
Annual Total MEI dose (rem)	3.4×10^{-4}
Total MEI LCF probability (unitless)	1.7×10^{-7}
Annual Total population dose (person-rem)	8.2
Total population LCFs (unitless)	0.004
Annual Collective worker dose (rem)	859
Collective worker LCFs (unitless)	0.34
Waste Generation (Life-Cycle Waste)	
High-level waste generation (cubic meters)	94,681
Low-level waste generation (cubic meters)	430,401
Hazardous/mixed waste generation (cubic meters)	14,745
Transuranic waste generation (cubic meters)	18,532
Utilities and Energy	
Annual electricity consumption (megawatt-hours)	5.77×10^5
Water usage (liters)	1.79×10^{10}

a. MEI = Maximally Exposed Individual.

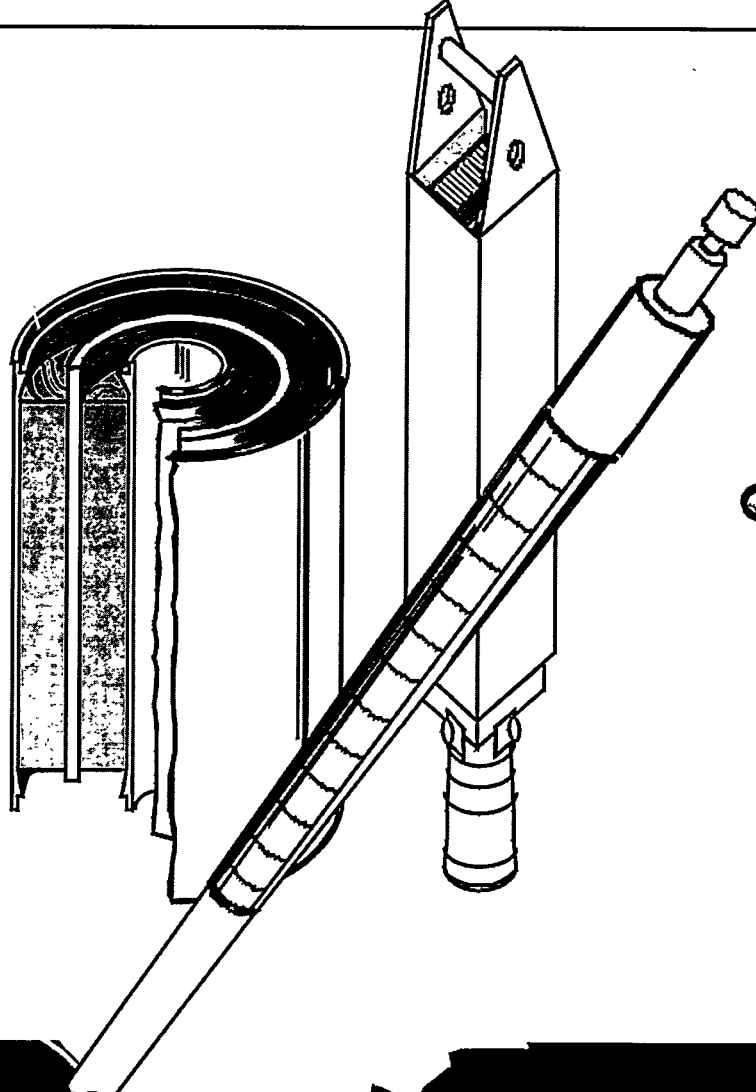
b. LCF = Latent Cancer Fatality.

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Savannah River Site

Spent Nuclear Fuel Management

FINAL ENVIRONMENTAL IMPACT STATEMENT



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Department of Energy • Savannah River Operations Office • Aiken, SC

COVER SHEET

RESPONSIBLE AGENCY: U.S. Department of Energy (DOE)

TITLE: Savannah River Site, Spent Nuclear Fuel Management Final Environmental Impact Statement (DOE/EIS-0279)

EC

CONTACT: For additional information on this environmental impact statement, write or call:

Andrew R. Grainger, NEPA Compliance Officer
U.S. Department of Energy, Savannah River Operations Office, Building 742A, Room 183
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The EIS is also available on the internet at: <http://tis.eh.doe.gov/nepa/docs/docs.htm>.

For general information on the process that DOE follows in complying with the National Environmental Policy Act, write or call:

EC

Ms. Carol M. Borgstrom, Director, Office of NEPA Policy and Assistance, EH-42
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ABSTRACT: The proposed DOE action considered in this environmental impact statement (EIS) is to implement appropriate processes for the safe and efficient management of spent nuclear fuel and targets at the Savannah River Site (SRS) in Aiken County, South Carolina, including placing these materials in forms suitable for ultimate disposition. Options to treat, package, and store this material are discussed. The material included in this EIS consists of approximately 68 metric tons heavy metal (MTHM) of spent nuclear fuel (20 MTHM of aluminum-based spent nuclear fuel at SRS, as much as 28 MTHM of aluminum-clad spent nuclear fuel from foreign and domestic research reactors to be shipped to SRS through 2035, and 20 MTHM of stainless-steel or zirconium-clad spent nuclear fuel and some Americium/Curium Targets stored at SRS.

EC

Alternatives considered in this EIS encompass a range of new packaging, new processing, and conventional processing technologies, as well as the No Action Alternative. A preferred alternative is identified in which DOE would prepare about 97 percent by volume (about 60 percent by mass) of the aluminum-based fuel for disposition using a melt and dilute treatment process. The remaining 3 percent by volume (about 40 percent by mass) would be managed using chemical separation. Impacts are assessed primarily in the areas of water resources, air resources, public and worker health, waste management, socioeconomic, and cumulative impacts.

PUBLIC INVOLVEMENT: DOE issued the Draft Spent Nuclear Fuel Management EIS on December 24, 1998, and held a formal public comment period on the EIS through February 8, 1999. In preparing the Final EIS, DOE considered comments received via mail, fax, electronic mail, and transcribed comments made at public hearings held in Columbia, S.C. on January 28, 1999, and North Augusta, S.C. on February 2, 1999. Completion of the Final EIS has been delayed because DOE has performed additional analyses of the melt and dilute technology, discussed in Chapter 2 and Appendix G. Comments received and DOE's responses to those comments are found in Appendix G of the EIS.

EC

FOREWORD

EC | The U.S. Department of Energy (DOE) published a Notice of Intent (NOI) to prepare this environmental impact statement (EIS) on December 31, 1996 (61 FR 69085). As described in the NOI, DOE's proposal in general terms is to implement appropriate actions to manage safely and efficiently spent nuclear fuel (SNF) and targets that are currently located or expected to be received at the Savannah River Site (SRS), including placing these materials in forms suitable for disposition. This EIS assesses the potential environmental impacts associated with storing, treating, and packaging these materials, including onsite transportation activities.

EC | The NOI requested public comments and suggestions for DOE to consider in its determination of the scope of the EIS, and announced a public scoping period that ended on March 3, 1997. DOE held a scoping meeting in North Augusta, South Carolina on January 30, 1997. During the scoping period, individuals, organizations, and government agencies submitted 118 comments that DOE considered applicable to the management of SNF at the SRS.

Transcripts of public testimony, copies of scoping letters, scoping comments and DOE responses to those comments, and reference materials cited in the EIS are available for review in the DOE Public Reading Room, University of South Carolina at Aiken, Gregg-Graniteville Library, University Parkway, Aiken, South Carolina.

TC | A Notice of Availability for the Draft EIS appeared in the *Federal Register* on December 24, 1998. Public meetings to discuss and receive comments on the Draft EIS were held on Thursday, January 28, 1999 in Columbia, S.C. and on Tuesday, February 2, 1999 in North Augusta, S.C. The public comment period ended on February 8, 1999. Comments and DOE responses to comments are in Appendix G.

Changes from the Draft EIS are indicated in this Final EIS by vertical change bars in the margin. In cases where changes were made in response to comments, the comment number (as listed in Appendix G) is listed next to the vertical change bar. Many of the technical changes are the result of the availability of updated information since publication of the Draft EIS.

TC

DOE has prepared this EIS in accordance with the National Environmental Policy Act (NEPA) regulations of the Council on Environmental Quality (40 CFR 1500-1508) and DOE NEPA Implementing Procedures (10 CFR 1021). This EIS identifies the methods used for analyses and the scientific and other sources of information consulted. In addition, it incorporates, directly or by reference, available results of ongoing studies. The organization of the EIS is as follows:

- Chapter 1 describes the purpose and need for SNF management at the SRS (i.e., to develop and implement a safe and efficient management strategy that includes preparing SNF for ultimate disposition), and describes the types of SNF to which the EIS applies.
- Chapter 2 identifies the alternatives that DOE is considering for management of SNF at the SRS.
- Chapter 3 describes the SRS environment as it relates to the alternatives described in Chapter 2.
- Chapter 4 assesses the potential environmental impacts of the alternatives for construction activities, normal operations, and accidents.
- Chapter 5 discusses the cumulative impacts of SNF management actions in relation to

- impacts of other past, present, and foreseeable future activities at the SRS.
- Chapter 6 identifies irreversible or irretrievable resource commitments.
 - Chapter 7 discusses regulatory requirements, including applicable statutes, DOE Orders, and state and Federal regulations.
 - Appendix A describes the technologies that DOE considered for implementing the SNF management alternatives described in Chapter 2.
 - Appendix B describes previously identified facility vulnerabilities specific to SRS SNF management, their recommended corrective actions, and the current status of those corrective actions.
 - Appendix C describes the SNF assigned to SRS for management and the categories into which DOE has grouped these fuels.
 - Appendix D provides detailed descriptions of accidents that could occur at SRS facilities during the management of SNF.
 - Appendix E describes assumed durations for each SNF management activity necessary to implement the alternatives described in Chapter 2.
 - Appendix F lists estimated incremental non-radiological air concentrations attributable to SNF management activities.
 - Appendix G describes public comments received on the Draft EIS and DOE responses.

TC

Change Bars

Changes from the Draft EIS are indicated in this Final EIS by vertical change bars in the margin. The bars are marked TC for technical changes, EC for editorial changes, or if the change was made in response to a public comment, the designated comment number is as listed in Appendix G of the EIS.

EC

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ACRONYMS, ABBREVIATIONS, AND USE OF SCIENTIFIC NOTATION

Acronyms

AAQS	ambient air quality standard
ALARA	as low as reasonably achievable
CFR	Code of Federal Regulations
CO	carbon monoxide
D&D	decontamination and decommissioning
DNFSB	Defense Nuclear Facilities Safety Board
DOE	U.S. Department of Energy
DWPF	Defense Waste Processing Facility
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
ES&H	environment, safety and health
FR	Federal Register
GMODS	glass material oxidation and dissolution system
HEPA	high-efficiency particulate air [filter]
HEU	highly enriched uranium
HLW	high-level waste
IMNM	Interim Management of Nuclear Material
INEEL	Idaho National Engineering and Environmental Laboratory
ISO	International Organization for Standardization
LCF	latent cancer fatality
LEU	low enriched uranium
MCL	maximum contaminant level
MEI	maximally exposed (offsite) individual
MTHM	metric tons of heavy metal

NAS	National Academy of Sciences
NCRP	National Council on Radiation Protection and Measurements
NESHAP	national emission standards for hazardous air pollutants
NIMS	nuclear incident monitoring system
NO _x	nitrogen oxides
NPDES	national pollutant discharge elimination system
NRC	Nuclear Regulatory Commission
O ₃	ozone
OSHA	Occupational Safety and Health Administration
PM ₁₀	particulate matter less than 10 microns in diameter
RBOF	Receiving Basin for Offsite Fuel
RINM	reactor irradiated nuclear materials
ROD	Record of Decision
SCDHEC	South Carolina Department of Health and Environmental Control
SMDF	Saltstone Manufacturing and Disposal Facility
SNF	spent nuclear fuel
SO ₂	sulfur dioxide
SRI	Savannah River Natural Resources Management and Research Institute
SRS	Savannah River Site
TRIGA	Training Research Isotope general atomic [spent fuel]
TSP	total suspended particulates
TSS	total suspended solids
VLEU	very low enriched uranium
WSRC	Westinghouse Savannah River Company

Abbreviations for Measurements

cfm	cubic feet per minute
cfs	cubic feet per second = 448.8 gallons per minute = 0.02832 cubic meter per second
cm	centimeter
gpm	gallons per minute
kg	kilogram
L	liter = 0.2642 gallon
lb	pound = 0.4536 kilogram
mg	milligram
μCi	microcurie
μg	microgram
pCi	picocurie
°C	degrees Celsius = $5/9$ (degrees Fahrenheit – 32)
°F	degrees Fahrenheit = $32 + 9/5$ (degrees Celsius)

Use of Scientific Notation

Very small and very large numbers are sometimes written using “scientific notation” or “E-notation” rather than as decimals or fractions. Both types of notation use exponents to indicate the power of 10 as a multiplier (i.e., 10^n , or the number 10 multiplied by itself “n” times; 10^{-n} , or the reciprocal of the number 10 multiplied by itself “n” times).

For example: $10^3 = 10 \times 10 \times 10 = 1,000$

$$10^{-3} = \frac{1}{10 \times 10 \times 10} = 0.001$$

In scientific notation, large numbers are written as a decimal between 1 and 10 multiplied by the appropriate power of 10:

4,900 is written $4.9 \times 10^3 = 4.9 \times 10 \times 10 \times 10 = 4.9 \times 1,000 = 4,900$

0.049 is written 4.9×10^{-2}

1,490,000 or 1.49 million is written 1.49×10^6

A positive exponent indicates a number larger than or equal to one, a negative exponent indicates number less than one.

In some cases, a slightly different notation (“E-notation”) is used, where “ $\times 10$ ” is replaced by “E” and the exponent is not superscripted. Using the above examples

$$4,900 = 4.9 \times 10^3 = 4.9\text{E}+03$$

$$0.049 = 4.9 \times 10^{-2} = 4.9\text{E}-02$$

$$1,490,000 = 1.49 \times 10^6 = 1.49\text{E}+06$$

*Acronyms, Abbreviations, and Use of Scientific Notation***Metric Conversion Chart**

To convert into metric			To convert out of metric		
If you know	Multiply by	To get	If you know	Multiply by	To get
Length					
inches	2.54	centimeters	centimeters	0.3937	inches
feet	30.48	centimeters	centimeters	0.0328	feet
feet	0.3048	meters	meters	3.281	feet
yards	0.9144	meters	meters	1.0936	yards
miles	1.60934	kilometers	kilometers	0.6214	miles
Area					
sq. inches	6.4516	Sq. centimeters	sq. centimeters	0.155	sq. inches
sq. feet	0.092903	sq. meters	sq. meters	10.7639	sq. feet
sq. yards	0.8361	sq. meters	sq. meters	1.196	sq. yards
acres	0.0040469	sq. kilometers	sq. kilometers	247.1	acres
sq. miles	2.58999	sq. kilometers	sq. kilometers	0.3861	sq. miles
Volume					
fluid ounces	29.574	milliliters	milliliters	0.0338	fluid ounces
gallons	3.7854	liters	liters	0.26417	gallons
cubic feet	0.028317	cubic meters	cubic meters	35.315	cubic feet
cubic yards	0.76455	cubic meters	cubic meters	1.308	cubic yards
Weight					
ounces	28.3495	grams	grams	0.03527	ounces
pounds	0.4536	kilograms	kilograms	2.2046	pounds
short tons	0.90718	metric tons	metric tons	1.1023	short tons
Temperature					
Fahrenheit	Subtract 32 then multiply by 5/9ths	Celsius	celsius	Multiply by 9/5ths, then add 32	Fahrenheit

Metric Prefixes

Prefix	Symbol	Multiplication Factor
exa-	E	1 000 000 000 000 000 000 = 10^{18}
peta-	P	1 000 000 000 000 000 = 10^{15}
tera-	T	1 000 000 000 000 = 10^{12}
giga-	G	1 000 000 000 = 10^9
mega-	M	1 000 000 = 10^6
kilo-	k	1 000 = 10^3
centi-	c	0.01 = 10^{-2}
milli-	m	0.001 = 10^{-3}
micro-	μ	0.000 001 = 10^{-6}
nano-	n	0.000 000 001 = 10^{-9}
pico-	p	0.000 000 000 001 = 10^{-12}
femto-	f	0.000 000 000 000 001 = 10^{-15}
atto-	a	0.000 000 000 000 000 001 = 10^{-18}

CHAPTER 1. INTRODUCTION

EC | The management of spent nuclear fuel (SNF) has been an integral part of the mission of the Savannah River Site (SRS) for more than 40 years. Until the early 1990s, SNF management consisted primarily of short-term onsite storage and processing in the SRS chemical separation facilities to produce strategic nuclear materials.

EC | With the end of the Cold War, the U.S. Department of Energy (DOE) decided to phase out processing of SNF for the production of nuclear weapons materials (DOE 1992). Therefore, the management strategy for this fuel has shifted from short-term storage and processing for the recovery of highly-enriched uranium and transuranic isotopes to stabilization, when necessary, and storage pending final disposition that includes preparing aluminum-based SNF for placement in any potential geologic repository. In addition to the fuel already onsite, the SRS will receive SNF from foreign research reactors until 2009 and from domestic research reactors until, potentially, 2035. As a result, the safe and efficient management of SNF will continue to be an important SRS mission.

TC |

This EIS evaluates the potential environmental impacts of DOE's proposed plans for managing SNF assigned to SRS.

1.1 Background

1.1.1 HISTORIC MISSIONS

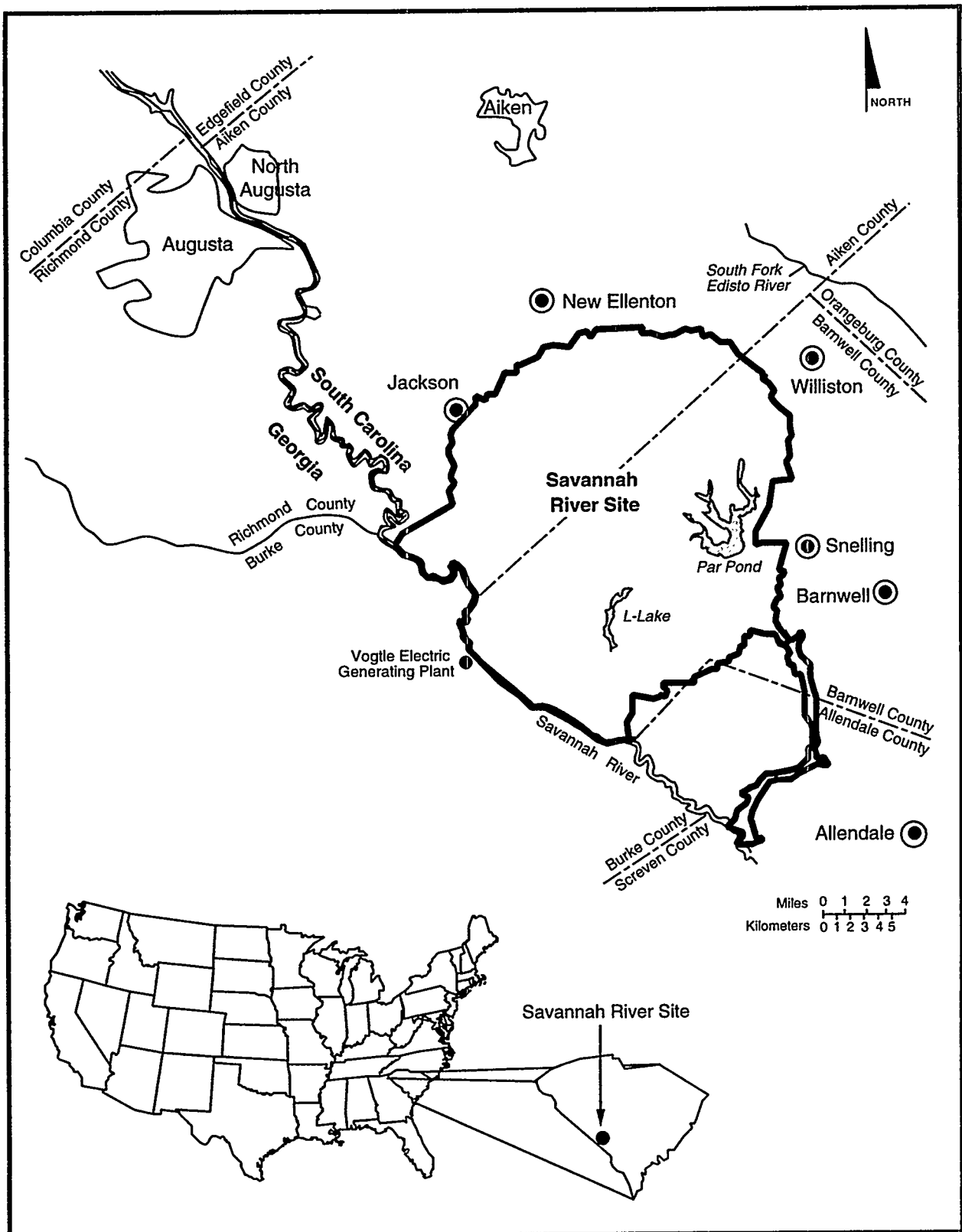
The U.S. Atomic Energy Commission, a DOE predecessor agency, established the SRS in the early 1950s. The Site occupies an area of approximately 300 square miles (800 square kilometers) adjacent to the Savannah River, primarily in Aiken and Barnwell Counties in South Carolina. It is approximately 25 miles (40 kilometers) southeast of Augusta, Georgia, and 20 miles (32 kilometers) south of Aiken, South Carolina (Figure 1-1).

For the past 40 years the SRS mission has been the production of special radioactive isotopes to support national programs. Historically, the primary Site mission was the production of strategic isotopes (plutonium-239 and tritium) for use in the development and production of nuclear weapons. The SRS produced other isotopes (e.g., californium-252, plutonium-238, americium-241) to support research in nuclear medicine, space exploration, and commercial applications. DOE produced these isotopes in the five SRS production reactors. After the material was produced at the SRS, it was shipped to other DOE sites for fabrication into desired forms.

1.1.2 FUEL CYCLE

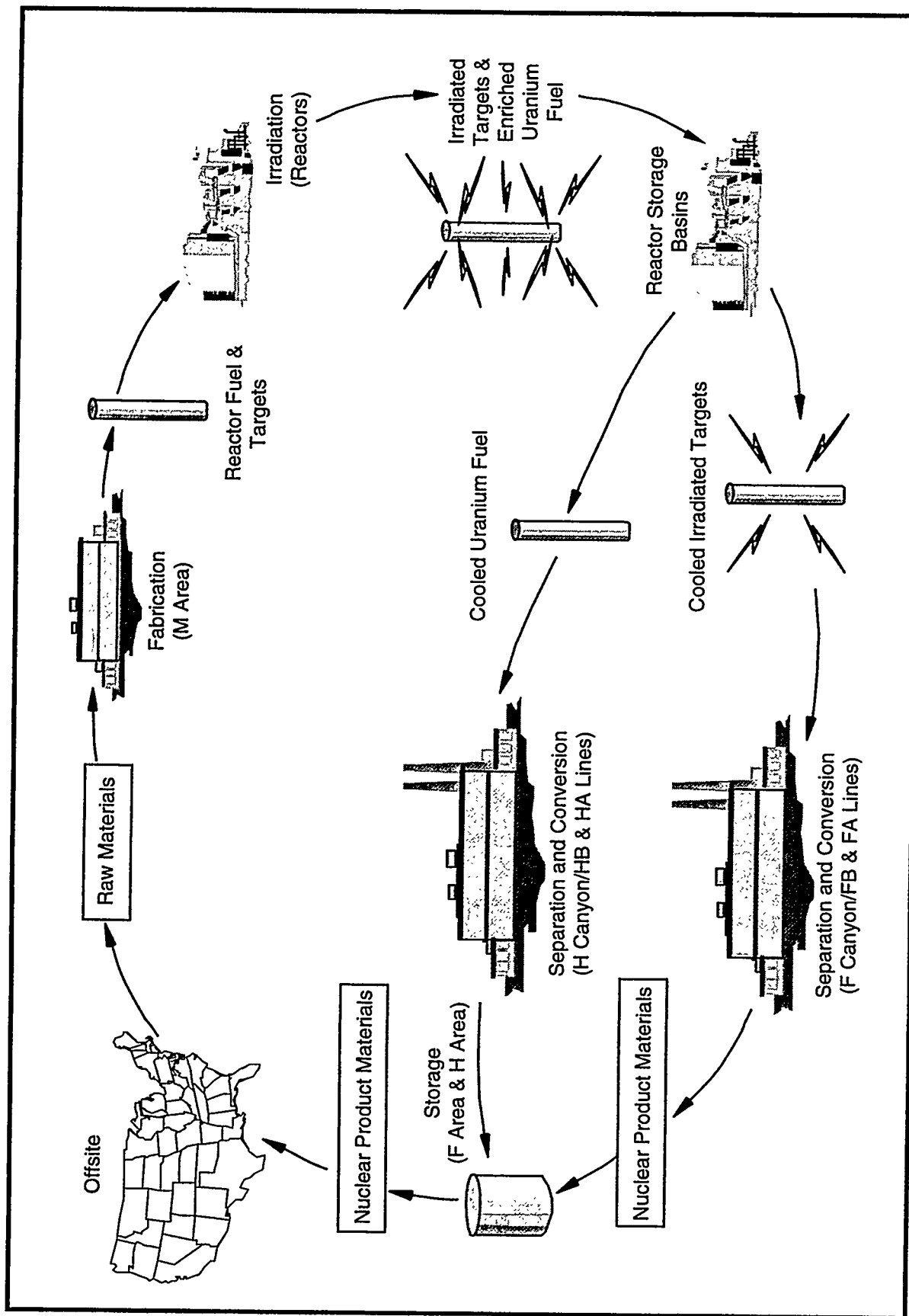
The material in the SRS reactors consisted of nuclear fuel and targets. The nuclear fuel was enriched uranium that was alloyed with aluminum and then clad with aluminum. The targets were either oxides or metallic forms of various isotopes such as neptunium-237 or uranium-238 that were clad with aluminum. Fuel and targets were fabricated at the SRS and placed in the reactors, and then the reactors operated to create the neutrons necessary to transmute the target material. For example, neptunium-237 targets were irradiated to produce plutonium-238, a material used by the National Aeronautics and Space Administration as a power source for deep space probes. After irradiation, the fuel and targets (collectively referred to as spent nuclear fuel) were removed from the reactors and placed in water-filled basins for short-term storage, about 12 to 18 months, before they were processed in the SRS separations facilities. Figure 1-2 shows the historic fuel and target cycle.

During processing, SNF was chemically dissolved in F or H Canyon to recover the uranium and transuranic isotopes. The recovered material was used in nuclear weapons programs or



NW SNF EIS/PubsOnly/SNF_Chap_1/Grlx_c1/11-1.ai

Figure 1-1. Location of the Savannah River Site.



NW SNF EIS/PubsOnly/SNF_Chap_1/Gfx_c1/f1-2.ai

Figure 1-2. Historic nuclear materials production cycle at the Savannah River Site.

for commercial applications. The remaining residue from the fuel, high-level radioactive waste consisting primarily of fission products and cladding in liquid form, was transferred to large steel tanks for storage. The high-level waste is currently being vitrified in the Defense Waste Processing Facility at the SRS to prepare it for disposal in any potential geologic repository.

1.1.3 CHANGING MISSIONS

With the end of the Cold War there was a decreased need for the strategic nuclear material that was produced at the SRS. In 1992, the Secretary of Energy directed that processing operations be phased out throughout the DOE complex, effectively halting the SRS mission to produce strategic nuclear materials such as plutonium-239. However, SNF and targets from previous production reactor irradiation cycles remained in storage at K-, L-, C-, and P-Reactor Disassembly Basins. (Chapter 2 describes SRS SNF storage facilities.)

In addition to nuclear material production missions, another mission for the SRS was (and continues to be) the receipt of SNF from DOE, domestic, and foreign research reactors. These reactors were operated by DOE, universities, and research institutions for educational and research purposes and to produce isotopes for nuclear medicine. Historically, SNF from these reactors was stored in the Receiving Basin for Offsite Fuel at SRS. In the past, much of the research reactor SNF was processed in the same manner as spent fuel from SRS production reactors. However, with the end of the Site's strategic nuclear materials production mission, SNF from research reactors has been accumulating in the Receiving Basin for Offsite Fuel and in the L-Reactor Disassembly Basin.

Some of the research reactor spent nuclear fuel sent to SRS was not aluminum based. Because DOE did not have the capability to process that type of SNF at SRS, it was placed in wet storage at the Receiving Basin for Offsite Fuel, where it remains in storage.

By 1995 DOE was storing about 195 metric tons heavy metal (MTHM [metric tons heavy metal] – the mass of uranium in the fuel or targets, excluding cladding, alloy materials, and structural materials) – of aluminum-based SNF in the SRS reactor disassembly basins and the Receiving Basin for Offsite Fuel. DOE also was storing about 20 MTHM of non-aluminum-based SNF in the Receiving Basin for Offsite Fuel.

1.1.4 STABILIZATION

DOE has taken action to stabilize about 175 MTHM of the 195 MTHM of aluminum-based SNF that was in storage at SRS in 1995. DOE decided to stabilize this material following completion of the *Interim Management of Nuclear Materials Environmental Impact Statement* (DOE 1995a). The primary purpose of the actions described in that environmental impact statement (EIS) was to correct or eliminate potential health and safety vulnerabilities related to some of the methods used to store nuclear materials (including SNF) at SRS. The vulnerable SNF had been stored in wet storage basins with poor water quality. The poor water quality resulted in corrosion and failure of the cladding on the fuel and subsequent releases of radioactive fission products to the water of the storage basins. In 1996, SRS began stabilizing vulnerable aluminum-based uranium metal SNF in F Canyon. That work is complete. Vulnerable aluminum-based SNF still is being stabilized in H Canyon and that work is expected to continue through 2002. In the *Interim Management of Nuclear Materials EIS* (DOE 1995a), DOE identified 20 MTHM (out of 195 MTHM) of aluminum-based SNF at SRS that was "stable," i.e., that likely could be safely stored for about 10 more years, pending decisions on final disposition. That 20 MTHM of aluminum-based SNF is included in this EIS.

1.1.5 SPENT NUCLEAR FUEL CONSOLIDATION

In May 1995, DOE decided (60 FR 28680) under the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environ-*

mental Restoration and Waste Management Programs Final Environmental Impact Statement to consolidate existing and newly generated SNF at three existing Departmental sites based on the fuel type, pending future decisions on ultimate disposition. Specifically, DOE decided that existing Hanford production reactor fuel would remain at Hanford, aluminum-based SNF (excluding the aluminum-based SNF at Hanford) would be consolidated at SRS, and non-aluminum-based SNF would be consolidated at the Idaho National Engineering and Environmental Laboratory (INEEL). DOE stated that decisions on preparing the SNF for final disposition would be made under site-specific National Environmental Policy Act evaluations. As a result of DOE's decision to consolidate SNF storage, DOE will transfer 20 MTHM of non-aluminum-based SNF from SRS to INEEL and will transfer about 5 MTHM of aluminum-based SNF at INEEL to SRS. DOE estimates these transfers could begin about 2009 and may be completed by 2017. Thus, the non-aluminum-based SNF at SRS and the aluminum-based SNF from INEEL that will be transferred to the SRS are included in this EIS. Additionally, as a result of the consolidation decision DOE reached under the *Programmatic Spent Nuclear Fuel Management and Idaho mental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995b), SRS could receive about 5 MTHM of aluminum-based SNF from domestic research reactors. Shipments from domestic research reactors could continue through 2035. Material expected to be received from domestic research reactors is included in this EIS.

In May 1996, DOE announced a decision (61 FR 25092) under the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (Nonproliferation Policy and Spent Fuel EIS) to accept about 18 MTHM of aluminum-based SNF containing uranium of United States origin from foreign research reactors for management in the United States at the SRS. The receipt of foreign research reactor SNF at SRS is now underway and receipts are scheduled to be completed by

2009. The 18 MTHM of foreign research reactor SNF that could be received at SRS is included in the scope of this EIS. (Recent decisions by some foreign research reactor operators have reduced the quantity of SNF expected to be shipped to SRS from about 18 MTHM to about 14 MTHM; however, the 18 MTHM projection is used for analysis purposes in this EIS because foreign research reactor operators still have the option to ship to the United States.)

1.1.6 PREPARATION FOR DISPOSITION

In summary, the total quantity of aluminum-based SNF at SRS that must be managed and prepared for disposition is as follows: 20 MTHM in existing SRS wet storage basins; about 10 MTHM to be received from INEEL and domestic research reactors; and about 18 MTHM to be received from foreign research reactors. Additionally, SRS must manage about 20 MTHM of non-aluminum-based SNF until it is transferred to INEEL.

1.2 Purpose and Need for Action

DOE anticipates placing most of its aluminum-based SNF inventory in a geologic repository after treatment or repackaging. However, DOE does not expect any geologic repository to be available until at least 2010 and shipments from DOE sites would not begin until about 2015. Until a repository is available, the Department intends to develop and implement a safe and efficient SNF management strategy that includes preparing aluminum-based SNF stored at SRS or expected to be shipped to SRS for disposition offsite. DOE is committed to avoiding indefinite storage at the SRS of this nuclear fuel in a form that is unsuitable for final disposition. Therefore, DOE needs to identify management technologies and facilities for storing and treating this SNF in preparation for final disposition.

1.3 Scope

This EIS evaluates potential environmental impacts from managing SNF that currently is located or expected to be located at SRS. The

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TC | evaluation includes impacts from the construction and operation of facilities (either new or modified existing facilities) that would be used to receive, store, treat, and package SNF in preparation for ultimate disposition. Onsite transportation impacts are considered, however, no impacts associated with transporting SNF to SRS are included, because these impacts have been covered in other EISs. The potential impacts of transporting SNF to a geologic repository are discussed (in Chapter 4) for completeness but no decisions related to transporting SNF offsite will be made under this EIS. Transportation of SNF (and high-level waste) to a federal repository will be addressed in the EIS for a federal repository (see Section 1.6). The Yucca Mountain EIS is being prepared as part of the process to determine whether to recommend the Yucca Mountain site as the site of the Nation's first geologic repository for SNF and high-level radioactive waste.

EC | In this EIS, DOE is evaluating the management of about 48 MTHM of aluminum-based SNF for treatment and storage (20 MTHM of aluminum-based SNF stored at SRS and about 28 MTHM of aluminum-based SNF from foreign and domestic research reactors that could be shipped to SRS until 2009 and from domestic research reactors that could be shipped to SRS until 2035).

TC | DOE also evaluates transferring 20 MTHM of non-aluminum-clad spent nuclear fuel currently stored in the Receiving Basin for Offsite Fuel at SRS to a new dry storage facility at SRS. This transfer would occur only if a dry storage facility were built as part of the implementation of a new treatment technology to prepare aluminum-based spent nuclear fuel for disposition (potential technologies are discussed in Section 2.2) and if the dry storage facility became operational before the non-aluminum-clad fuel was transferred to the INEEL. The transfer to dry storage would occur after the fuel had been relocated from the Receiving Basin for Offsite Fuel to the L-Reactor Disassembly Basin in support of activities necessary to phase out the use of the Receiving Basin for Offsite Fuel by fiscal year 2007.

This EIS does not evaluate the impacts of managing the non-aluminum-clad fuel at INEEL or of transporting the fuel to INEEL. These impacts were documented in the SNF programmatic EIS (PEIS) (DOE 1995b) and were evaluated as part of the process DOE used to decide to consolidate the storage of non aluminum-clad spent nuclear fuel at the INEEL.

SRS is storing Mark-51 and other targets in the Receiving Basin for Offsite Fuel (RBOF) in the Site's H-Area. This EIS evaluates the impacts of continuing to store the Mark-51 and other targets in RBOF, and evaluates an alternative of transferring them to dry storage to provide flexibility in material management operations.

DOE is evaluating potential uses for this material and the operations and facilities that would be necessary. The Mark-51 and other targets (described in Section 1.5 of this EIS) contain americium and curium isotopes that could be used to produce elements with higher atomic numbers such as californium-252. Californium-252 is used as a neutron source for radiography and in the treatment of certain types of cancer and for research in basic chemistry, nuclear physics, and solid-state chemistry. If DOE were to determine that a programmatic need for this material exists, the targets would continue to be stored at the SRS pending preparations to ship them to another DOE facility where isotope production capability currently exists or could be constructed and operated. SRS does not have isotope production capability.

This EIS does not evaluate the impacts of utilizing target material for programmatic purposes such as production of californium. DOE would perform the appropriate National Environmental Policy Act review to evaluate the impacts of shipment of the targets to an isotope production facility and of construction (or modification) and operation of the production facility, should such a programmatic purpose be identified.

DOE is storing the Mark-18 targets in wet basins at the SRS. These targets are similar to the Mark-51 and other targets in that they contain americium and curium that could be used to

produce elements with higher atomic numbers such as californium-252. They are different from the small (about two feet in length) Mark-51 and other targets because the Mark 18s are about 12 feet long and therefore have different requirements for storage, transportation and use. As is the case with the Mark-51 and other targets, DOE is not proposing any actions that would lead to programmatic use of the Mark-18 targets at this time. Because of their length, the Mark-18 targets would have to be reduced in size for use in production facilities at another DOE facility or transfer to dry storage at the SRS. This EIS considers only continued wet storage of Mark-18 targets. However, the Interim Management of Nuclear Materials EIS (which is incorporated herein by reference) considered the alternative of processing the Mark-18 targets in the SRS canyons, should they present potential health and safety vulnerabilities. See Section 1.5 of this EIS for more information.

1.4 Decisions to be Based on this EIS

DOE expects to make the following decisions on the management and preparation of SNF for storage and ultimate disposition.

- The selection of the appropriate treatment or packaging technologies to prepare aluminum-based SNF that is to be managed at SRS.
- Whether DOE should construct new facilities or use existing facilities to store and treat, or package aluminum-based SNF that is expected to be managed at SRS.
- Whether DOE should repackage and dry-store stainless-steel and zirconium-clad SNF pending shipment to the Idaho National Engineering and Environmental Laboratory.

- Whether DOE should repackage and dry-store Mark-51s and other americium/curium targets in the event dry-storage capability becomes available at SRS.

1.5 Spent Nuclear Fuel Groups

This section introduces the basic terminology for describing SNF and provides more information on the approximately 68 MTHM of SNF subject to analysis in this EIS.

DOE has categorized the spent fuel considered in this EIS into six groups (Group A through Group F). The categorization is based on such characteristics as fuel size, physical or chemical properties, or radionuclide inventories. DOE grouped the fuel to distinguish how it could apply the management alternatives evaluated in the EIS (Section 2.2). Table 1-1 lists the fuel groups and the amount of fuel in each group. Appendix C provides more detailed information regarding fuel types, quantities, locations, radionuclide inventories, and curie content.

The aluminum-based fuels currently stored at SRS include some fuels that were not originally aluminum-clad (EBR-II and Sodium Breeder Experimental Reactor Fuel). Additionally, the aluminum-based category consists of one element not yet received but due to be shipped to SRS (the Advanced Reactivity Measurement Facility Core Filter Block). Most of the fuels that were not originally aluminum-clad (but are included under this EIS's major category of aluminum-based fuel) have been declad and placed in aluminum cans. In their present form they can be processed at the SRS through the existing technologies on site. Other fuels at SRS which are non-aluminum-clad fuels cannot be processed in their existing form using the existing technologies and are characterized in this EIS as non-aluminum-based fuel. The Core Filter Block is included under the category of

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Table 1-1. Spent nuclear fuel groups.

Fuel group	Volume (MTRE) ^a	Mass (MTHM) ^b	
A. Uranium and Thorium Metal Fuels	610	19	EC
B. Material Test Reactor-Like Fuels	30,800	20	TC
C. HEU/LEU ^c Oxides and Silicides Requiring Resizing or Special Packaging	470 ^d	8	
D. Loose Uranium Oxide in Cans	NA	0.7	TC
E. Higher Actinide Targets	NA	<0.1	
F. Non-Aluminum-Clad Fuels ^e	<u>1,900</u>	<u>20.4</u>	
Total	33,780	68.2	

NA = Not applicable

a. MTRE = Materials test reactor equivalent. An MTRE is a qualitative estimate of SNF volume that provides information on the amount of space needed for storage. An MTRE of Materials Test Reactor-Like Fuels would usually be one fuel assembly measuring about 3 inches by 3 inches by 2 feet long.

b. MTHM = Metric tons of heavy metal.

c. HEU = highly enriched uranium; LEU = low enriched uranium.

d. Fuel group also includes about 2,800 pins, pin bundles, and pin assemblies.

e. This fuel group will be shipped to Idaho National Engineering and Environmental Laboratory. It will not be treated at SRS.

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aluminum-based fuel since the most practical way of dealing with it (based on its unique configuration) is to process it utilizing the existing technology at SRS.

Uranium and Thorium Metal Fuels (Group A):

This group consists of fuels from the Experimental Breeder Reactor-II and the Sodium Reactor Experiment, as well as a core filter block from the Advanced Reactivity Measurement Facility at INEEL (that is scheduled to be transferred to SRS). This group also includes unirradiated Mark-42 targets that were manufactured from plutonium oxide-aluminum powder metal and formed into tubes that were clad with aluminum

The Experimental Breeder Reactor-II fuel and Sodium Reactor Experiment fuel are uranium metal that has been declad and stored in canisters in the Receiving Basin for Offsite Fuel. The declad fuel presents a potential health and safety vulnerability. These fuels have cores of reactive metals that were exposed when the fuel cladding was removed. Any contact of the reactive metal core with water would lead to relatively rapid oxidation of the core and

disintegration of the fuel. Should the existing storage containers leak, the metal fuel would corrode and release fission products to the water of the storage basin. Once the metal of the fuel is wetted, simply repackaging the fuel in a water-tight container would not arrest the corrosion and, in fact, could exacerbate storage concerns since potentially explosive hydrogen gas would continue to be generated inside the storage canister as the fuel continued to corrode. Water intrusion and subsequent fuel corrosion has already occurred with one Experimental Breeder Reactor-II canister stored in the Receiving Basin for Offsite Fuel. That material was processed in F Canyon to eliminate the problem. In the event that leaks were detected in any additional canisters prior to processing/treatment in accordance with decisions reached under this EIS, DOE would process those canisters in an SRS canyon facility. This management approach is consistent with the Records of Decision reached under the *Interim Management of Nuclear Materials Final Environmental Impact Statement* for other uranium metal SNF stored in the Receiving Basin for Offsite Fuel at the SRS. The *Interim Management of Nuclear Materials EIS* deferred decisions on the materials that did not pose immediate health and safety vulnerabilities

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because they were considered to be stable for 10 years and DOE wanted to provide the public an opportunity to comment as part of the overall planning for SNF at SRS.

The unirradiated Mark-42 targets were manufactured from plutonium oxide-aluminum powder metal and formed into tubes that were clad with aluminum. The plutonium oxide and aluminum were pressed together in the manufacturing process. As a result, the unirradiated targets are less durable than uranium-aluminum alloy SNF because of the particulate nature of the plutonium oxide but more durable (i.e., less reactive) than uranium metal SNF since the plutonium is already in oxide form. The unirradiated Mark-42 targets present a potential safety and health vulnerability in that should the cladding of these targets be breached, the plutonium oxide could migrate to the water of the storage basin.

The core filter block at INEEL is made of depleted uranium and was used as a neutron "filter" for reactivity experiments. As a result, the filter was subject to relatively short (or low-power level) exposure times in the test reactor and is only slightly irradiated. The core filter block contains cylindrical sleeves of various corrosion resistant metals at different diameters within the filter block.

DOE is unaware of any health or safety concerns related to the core filter block. The core filter block is a unique assembly in that it includes materials that would not be compatible with the melt and dilute process for aluminum-based SNF. Additionally, the core filter block is composed mainly of depleted uranium and has been exposed to relatively low power so it contains very little fissile material or fission products. Processing would not extend the time for planned canyon operations, would not generate recovered fissile material, and would produce only a few kilograms of depleted uranium.

There is uncertainty regarding the acceptability of the material in this fuel group in its current form into a repository due to the reactive nature of uranium metal or the particulate nature of

some of the material. The oxidation or burning of the metal in the repository could cause damage and spread radioactive particles throughout the repository. Although somewhat less reactive than pure metals, the uranium and thorium metal fuels discussed in this EIS (Group A) would need special attention to mitigate their reactivity.

This group accounts for approximately 2.0 percent of the volume of aluminum-based fuel that DOE is likely to manage at the SRS from now until 2035. Because the fuel in Group A is made of unalloyed metal (i.e., it contains little or no aluminum), it is more dense than most of the other spent fuel considered in this EIS. As a result, this small volume of fuel contains about 40 percent of the mass of heavy metal.

Materials Test Reactor-Like Fuels (Group B):

This group consists primarily of Materials Test Reactor fuels and other fuels of similar size and composition. Most research reactors – foreign and domestic – use Materials Test Reactor fuel, which has a flat or curved plate design. Figure 1-3 shows a typical Materials Test Reactor fuel assembly. Although these fuels come in a variety of shapes and compositions, the active fuel region is typically about 2 feet (0.6 meter) long and the overall assembly is about 4 feet (1.2 meters) long. The cross-section of an assembly is approximately square, about 3 inches (8 centimeters) on a side.

These fuels vary in enrichment. Approximately 70 percent of the Group B assemblies are highly enriched uranium, and the remainder are low enriched uranium. They are uranium-aluminum, uranium oxide-aluminum, or uranium silicide-aluminum alloy; all types are clad with aluminum. Group B accounts for approximately 97 percent of the volume of aluminum-based SNF that DOE will manage at SRS between now and 2035. DOE considers that there are no currently known health and safety vulnerabilities for this material that would preclude wet storage pending the operation of a new treatment technology.

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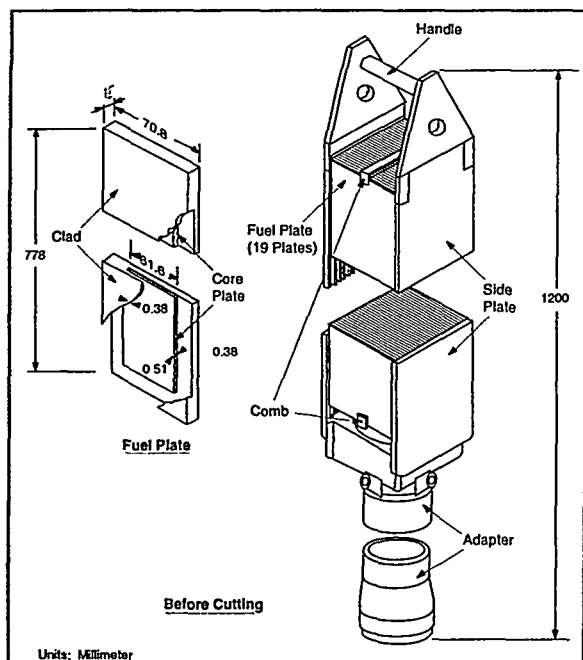


Figure 1-3. Typical Materials Test Reactor fuel assembly.

Although some Group B fuels are stored at SRS in the Receiving Basin for Offsite Fuel or in L Disassembly Basin, at present most are at domestic universities, foreign research reactors, and DOE research facilities pending shipment to the Site. All of the Group B fuels that are currently stored at SRS are "intact." The good condition of the cladding and the durability of the alloyed fuel at SRS provide a high degree of confidence that the fuel will not degrade during storage and that actions to correct potential health and safety vulnerabilities will not be necessary before treatment using the technology that DOE proposes to select under the record of decision from this EIS. DOE expects this will be true for most of the foreign and domestic research reactor SNF included in Group B that is yet to be shipped to SRS. However, if DOE determines that any of the Group B fuel presents a health and safety vulnerability, DOE would evaluate the situation and take appropriate action that could include canning the problem fuel or processing the fuel in one of the SRS canyon facilities. This management approach is consistent with the Record of Decision reached under the *Environmental Impact Statement on a*

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel.

HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging (Group C):

Fuels in this group are similar in composition to Group B fuels in that they are aluminum-based, highly enriched uranium (HEU) and low enriched uranium (LEU) oxides and silicides, but their size or shape might preclude packaging them in the disposal canisters proposed for use in a repository without resizing or special packaging considerations. Some fuel in this group is smaller in diameter and longer than Group B fuels or is larger than Group B fuels in both diameter and length; it often comes in odd shapes such as a 1.5-foot by 3-foot (0.46-meter by 0.9-meter) cylinder or a sphere with a diameter of 29 inches (74 centimeters). DOE would have to disassemble or use other volume-reduction activities to place such fuels in a nominal 17-inch direct co-disposal canister (see Section 2.2). At present, much of this fuel is at other DOE sites and in other countries but is scheduled to be received at SRS.

DOE expects that most of the fuel in this category is intact and would be managed as described above for Group B fuels. However, a small amount is not intact. That material consists of some fuel and one target that were cut or sectioned for research purposes. After the research was completed, the fuel and target pieces were canned in 14 cans and placed in wet storage. The origin and location of this material is discussed in Appendix C, Table C-3. The sectioned fuel and target present a potential health and safety vulnerability similar to that of the Group A fuel discussed previously. If a storage can were to leak, DOE would address the problem as described for the Group A fuel to prevent the release of fission products and particulate material to the water of a storage basin. Additionally, the current form of the fuel (i.e., failed) may not be acceptable in a repository because its integrity has been compromised.

Together Group B and Group C fuels represent 97 percent of all fuel to be managed at SRS, and 93 percent of the total fuel at SRS (including Group F fuels which will be shipped to Idaho National Engineering and Environmental Laboratory without treatment at SRS).

Loose Uranium Oxide in Cans (Group D):

This group consists of loose uranium oxide with fission products distributed through the material that has been stored in aluminum cans. This material, in its current particulate form, probably would not be acceptable for disposal in a repository because it is not in a tightly bound metal or ceramic matrix. Therefore, this group probably would require special packaging and/or treatment. Group D fuels also include targets in foreign countries that are liquid and that DOE expects would be converted to oxide prior to shipment to SRS. Only about 10 percent of the Group D fuel is in storage at SRS. The rest of the material has yet to be produced via foreign research reactor operations. Although eligible for shipment, most of this fuel is not part of the current shipping plan as projected by foreign research reactor operators.

The Group D fuel currently stored at SRS (676 cans of Sterling Forest Oxide fuel from the former medical isotope – production reactor; see Table C-4) presents a potential health and safety vulnerability similar to that of the Group A fuels. If a storage can leaked, DOE would address the problem as described for the Group A fuels to prevent the release of fission products and particulate matter to the water of an SRS storage basin. Group D comprises approximately 6 percent of the volume of the aluminum-based SNF that DOE could manage at SRS from now until 2035.

Higher Actinide Targets (Group E):

This group contains irradiated and unirradiated target materials used to generate radionuclides with atomic numbers higher than that of uranium. This material could be used to support such national programs as space exploration or medical research. The targets are aluminum-

clad plutonium oxide that contain significant quantities of americium and curium, which react under neutron irradiation to produce elements with still higher atomic numbers such as californium. All materials in this group are stored in the Receiving Basin for Offsite Fuel. Group E accounts for less than 1 percent of the volume of aluminum-based SNF DOE could manage at SRS from now until 2035.

The Higher Actinide Target fuel group consists of 60 Mark-51 targets, 114 other targets, and 65 Mark-18 targets. This material was evaluated in the *Final Environmental Impact Statement for Interim Management of Nuclear Materials*, (DOE/EIS-0220) and DOE decided the targets should remain in wet storage. In this EIS, DOE evaluates the continued wet storage of the Mark-51 and other targets pending shipment offsite. DOE also evaluates repackaging the Mark-51 and other targets to place them in a new dry storage facility so that the material could be transferred to dry storage if necessary to provide flexibility in spent fuel storage operations.

The Mark-18 targets are different from the Mark-51 and other targets in several ways. The most important distinction is that each Mark-18 target is one continuous piece about 12 feet long. The Mark-51 and other targets are about 2 feet long. The Mark-51 and other targets could be handled, transported and stored (including in a dry storage facility) in their current configuration. The 12-foot long Mark-18 targets would require size reduction for transport or storage in a dry storage facility. The standard method to reduce the size of the Mark-18 targets would be to cut them up under water in an SRS wet storage basin. The condition of the Mark-18 targets presents a health and safety vulnerability for under water cutting because of the suspected brittle condition of the targets and the uncertainty of the region of the target assemblies that contains the target product (i.e., americium and curium) and fission products. The brittle condition is due to a very long irradiation cycle in a reactor at the SRS. Cutting the targets using the existing site capability could result in the uncontrolled release of radioactive material to the water of the Receiving

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TC | Basin for Offsite Fuel. For these reasons, a previous DOE assessment of this material (see Section 1.6.2) concluded that the Department should consider processing the Mark-18 targets in F Canyon. Analysis of such alternatives are not included in this EIS because DOE performed that evaluation in the *Final Environmental Impact Statement for Interim Management of Nuclear Materials*, which is incorporated herein by reference. Those alternatives included dissolving the targets in F-Canyon and then vitrifying the americium and curium in a new F-Canyon vitrification facility, dissolving the targets in F-Canyon and recovering the americium and curium as an oxide, and dissolving the targets and transferring the americium and curium to the high-level waste tanks at the SRS.

Non-Aluminum-Clad Fuels (Group F):

This group consists of the large variety of stainless-steel or zirconium-clad SNF at SRS that DOE plans to ship to INEEL in accordance with decisions DOE reached under the SNF PEIS (DOE 1995b).

1.5.1 COMPARISON OF SPENT NUCLEAR FUEL GROUPS

TC | A comment was made regarding the differences between the fuel categories used in this EIS and the EIS for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada (i.e., Yucca Mountain EIS). The Notice of Availability of the Yucca Mountain Draft EIS was published on August 13, 1999 (64 FR 44217) and analyzes the options being considered for siting of a repository for spent nuclear fuel and high level waste.

Table 1-2 shows the categories being used in both EISs. The Yucca Mountain categories and MTHM numbers encompass fuel and targets being managed by SRS in preparation for ultimate disposition. Should a repository be developed, that fuel and most targets would be shipped, in one form or another, to the repository for ultimate disposition. Category F fuel will be shipped from SRS to INEEL under the Record of Decision for the Final Programmatic Spent Nuclear Fuel and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs EIS. As such, INEEL will be responsible for determining the ultimate disposition of category F fuel. Therefore, the 20.4 MTHM of non-aluminum clad fuel is not included in the Yucca Mountain categories for SRS managed fuel.

Category A is made up of 17 MTHM EBR-II (matching Yucca Mountain EIS category 1) and 2 MTHM SRE ("Thorium" part). The SRE is contained within Yucca mountain category 16.

Material within groups B and C of the SRS SNF EIS are included in groups 5, 6, and 7 of the Yucca Mountain EIS. Material within groups D & E of the SNF EIS are included in group 16 of the Yucca Mountain EIS. The material is made up of foreign research reactor and domestic research reactor fuel and targets and other target material produced at SRS.

Excluding group F, there is a 4.0 MTHM difference between the totals calculated for the SNF EIS table (47.8 MTHM) and the Yucca Mountain table (43.8 MTHM). The differences are due to recent decisions by some foreign research reactor (FRR) operators which have reduced the quantity of SNF expected to be shipped to SRS. However, the SRS SNF EIS uses the larger projected number because those FRRs still have the option to ship to the United States.

Table 1-2. Comparison of Spent Nuclear Fuel Groups.

NEPA document		Fuel group	Mass (MTHM) ^a
Savannah River Site Spent Nuclear Fuel Management EIS (DOE/EIS-0279)	A	Uranium and Thorium Metal Fuels	19
	B	Material Test Reactor-Like Fuels	20
	C	HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	8
	D	Loose Uranium Oxide	0.7
	E	Higher Actinide Targets	0.1
	F	Non-Aluminum-Clad Fuels	20.4
Draft EIS for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada (DOE/EIS-0250D) ^b	1	Uranium Metal	17
	5	Uranium Oxide, Failed/ Declad/ Aluminum Clad	3.2
	6	Uranium-Aluminide	8.7
	7	Uranium-Silicide	12
	16	Miscellaneous	2.9

a. MTHM = Metric tons of heavy metal.

b. Includes only Savannah River Site Fuel

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1.6 Relevant Documents

1.6.1 NATIONAL ENVIRONMENTAL POLICY ACT DOCUMENTS

Final Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement

DOE prepared this EIS (DOE 1995b) in compliance with a Court Order dated December 22, 1993, in the case of Public Service Company of Colorado v. Andrus, No. 91-0054-5-HLR (D. Idaho). The preferred alternative in the Final EIS, which DOE issued in April 1995, is Regionalization by Fuel Type. Volume 1 of this EIS analyzes at a programmatic level potential environmental impacts over the next 40 years of alternatives related to the transportation, receipt, processing, and storage of DOE-owned SNF. Volume 1 supports programmatic decisions on sites at which DOE will manage various types of SNF.

In the Record of Decision, which selected the preferred alternative for implementation (60 FR 28680), DOE decided to manage its SNF by type (fuel cladding and matrix material) at the

Hanford Site, the Idaho National Engineering and Environmental Laboratory, and the SRS. Section C.1.2 in Appendix C of this SRS SNF Management EIS discusses its relationship to the programmatic SNF EIS.

An amendment to the Record of Decision (61 FR 9441) reflects the October 16, 1995, Settlement Agreement between DOE, the State of Idaho, and the Department of the Navy by reducing the number of proposed spent fuel shipments to Idaho.

Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor SNF

This EIS (DOE 1996a) analyzes the management of foreign research reactor SNF that contains uranium originally produced or enriched in the United States. It also analyzes appropriate ways to manage such fuel received in the United States, amounts of fuel, shippers, periods of time over which DOE would manage the fuel, modes of transportation, and ownership of the fuel. In its Record of Decision (61 FR 25091), DOE stated it would accept from 41 listed countries aluminum-based spent fuel, Training Research Isotope General Atomic (TRIGA)

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spent fuel, and target material containing uranium enriched in the United States.

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Over the life of the foreign research reactor SNF acceptance program, DOE could accept approximately 19.2 MTHM of foreign research reactor SNF in as many as 22,700 separate elements and approximately 0.6 MTHM of target material. Most of the fuel will arrive through the Charleston Naval Weapons Station in South Carolina (about 80 percent), with a very limited amount arriving through the Concord Naval Weapons Station in California (about 5 percent). Most of the target material and some of the fuel (about 15 percent) will arrive overland from Canada. Shipments through Charleston began in September 1996 and those through Concord began in July 1998.

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After a limited period of storage, DOE will process and package the fuel as necessary at the SRS and the Idaho National Engineering and Environmental Laboratory to prepare it for disposal in a geologic repository. Section C.1.2 in Appendix C explains the relationship of the Foreign Research Reactor SNF EIS to this EIS.

Final Environmental Impact Statement Interim Management of Nuclear Materials

This EIS (DOE 1995a) evaluates actions to stabilize SRS materials that represent environmental, safety, and health vulnerabilities in their current storage condition or that might represent a vulnerability within the next 10 years.

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DOE has published four decisions under this EIS. In the first (60 FR 65300), DOE decided to process plutonium-242 solutions to oxide; vitrify americium and curium solutions to glass; blend highly-enriched uranium solutions down to low enrichment; process the plutonium in Mark-31 target slugs; process plutonium and uranium material in vaults to metal, oxide, or glass, if necessary; and process failed Taiwan Research Reactor SNF and a failed canister of Experimental Breeder Reactor-II SNF.

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DOE decided that processing the EBR-II fuel in unbreached canisters was not immediately nec-

essary. EBR-II fuel is clad and reactive, but only when it is in contact with water. The fuel inside a storage canister will not corrode as long as the canister retains its integrity. A monitoring and inspection program is in place that would detect any change in the integrity of the storage canisters. Any canisters that failed would be detected and the fuel then processed under the provisions of the Record of Decision to stabilize the material. This monitoring and inspection program applies as well to other fuel types in storage.

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In the first supplement to the Record of Decision (61 FR 6633), DOE decided to stabilize Mark-16 and -22 fuels by processing them in the SRS canyons and blending the resulting highly enriched uranium down to low enriched uranium; and to stabilize "other aluminum-clad targets" by dissolving them in the canyons. DOE will transfer the resulting nuclear material from the targets to the SRS high-level waste tanks for vitrification in the Defense Waste Processing Facility.

The second supplement to the Record of Decision (61 FR 48474) contains decisions on vitrifying neptunium-237 solutions, and on the stabilization of plutonium-239 solutions by converting them to a metal using the F and H Canyons and FB-Line.

In the third supplement to the Record of Decision (62 FR 17790), DOE decided to use the F Canyon and FB-Line to stabilize the remaining Taiwan Research Reactor SNF in the Receiving Basin for Offsite Fuel. These actions are relevant to the cumulative impacts assessment in this EIS (see Chapter 5).

Disposition of Surplus Highly Enriched Uranium Environmental Impact Statement

DOE prepared this EIS (DOE 1996b) because of the need to reduce the threat of nuclear weapons proliferation worldwide in an environmentally safe manner by reducing stockpiles of weapons-usable fissile materials, setting a non-proliferation example for other nations, and al-

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lowing peaceful, beneficial use of the material to the extent practical.

In the Record of Decision (61 FR 40619), DOE stated it would implement a program that will gradually blend as much as 85 percent of the surplus highly enriched uranium to a uranium-235 enrichment level of approximately 4 percent, and will blend the remaining surplus highly enriched uranium down to an enrichment level of about 0.9 percent for disposal as low-level waste. This will occur over 15 to 20 years. DOE could use different technologies at four potential blending facilities, including SRS and the Oak Ridge Reservation. Blending down of highly-enriched uranium would affect SRS operations and waste generation. This activity is relevant to the assessment of cumulative impacts (see Chapter 5).

Storage And Disposition Of Weapons-Usable Fissile Materials Programmatic Environmental Impact Statement

DOE prepared this programmatic EIS (DOE 1996c) to evaluate a safe and secure strategy for the long-term storage of weapons-usable fissile materials, primarily plutonium-239 and highly enriched uranium, and the disposition of weapons-usable plutonium that was surplus to national defense needs. This EIS included the SRS inventory of plutonium-239, highly enriched uranium, and other weapons-usable materials.

The Record of Decision (62 FR 3014) specified that DOE will expand or upgrade SRS facilities (i.e., the Actinide Packaging and Storage Facility) to consolidate weapons-usable plutonium, and will move plutonium pits now stored at the Rocky Flats Environmental Technology Site in Colorado to the Pantex Plant in Texas and non-pit plutonium materials to SRS. DOE will ship the non-pit plutonium to SRS only if a subsequent decision calls for the immobilization of plutonium at the Site. The DOE disposition strategy enables the immobilization of surplus plutonium in glass or ceramic material for disposal in a geologic repository, and the burning of some surplus plutonium as mixed oxide fuel

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in domestic commercial reactors with subsequent disposal of the spent fuel in a geologic repository in accordance with the Nuclear Waste Policy Act.

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DOE specified that it will determine the exact locations for disposition of these materials in site-specific EISs and in cost, technical, and nonproliferation studies. However, DOE has decided that it will locate a vitrification or immobilization facility (with a plutonium conversion facility) at either the Hanford Site in Washington or SRS, and that SRS is a candidate site for a potential mixed oxide fuel fabrication facility and a pit disassembly and conversion facility. The implementation of these decisions will require several years. The Programmatic Weapons-Usable Fissile Materials EIS is also relevant in the assessment of cumulative impacts that could occur at the SRS (see Chapter 5).

The Department issued an Amended Record of Decision (63 FR 43386) to the environmental impact statement, *Storage and Disposition of Weapons-Usable Fissile Materials*, on August 6, 1998. In order to support the early closure of the Rocky Flats Environmental Technology Site (RFETS) and the early deactivation of plutonium storage facilities at the Hanford Site, DOE modified, contingent upon the satisfaction of certain conditions, some of the decisions made in its Storage and Disposition ROD associated with surplus plutonium storage pending disposition. Namely, DOE will take steps that allow: (1) the accelerated shipment of all non-pit surplus weapons-usable plutonium from the RFETS (about 7 metric tons) to the SRS beginning in about 2000, in advance of completion of the Actinide Packaging and Storage Facility in 2001, and (2) relocation of all Hanford surplus weapons-usable plutonium (about 6.4 metric tons) to the SRS, between about 2002 and 2005, pending disposition. However, consistent with the Storage and Disposition PEIS ROD, DOE will only implement the movement of the RFETS and Hanford plutonium inventories to the SRS if the SRS is selected as the immobilization disposition site. DOE is preparing the *Surplus Plutonium Disposition EIS*, draft issued July 1998, as part of the decision-making proc-

ess for determining the immobilization site. The action described in this EIS is relevant in the assessment of cumulative impacts that could occur at SRS (see Chapter 5).

Final Defense Waste Processing Facility Supplemental Environmental Impact Statement

DOE prepared a Supplemental EIS to examine the impacts of completing construction and operating the Defense Waste Processing Facility at the SRS. This document (DOE 1994) assisted the Department in deciding whether and how to proceed with the Defense Waste Processing Facility project, given the changes to processes and facilities that had occurred since 1982, when it issued the original Defense Waste Processing Facility EIS. The Record of Decision (60 FR 18589) announced that DOE would complete the construction and startup testing of the Defense Waste Processing Facility, and would operate the facility using the In-Tank Precipitation process after the satisfactory completion of startup tests.

The alternatives evaluated in this EIS on the management of SNF could generate radioactive waste that DOE would have to handle or treat at facilities described in the Defense Waste Processing Facility Supplemental EIS and the SRS Waste Management EIS (see next paragraph). The Defense Waste Processing Facility Supplemental EIS is also relevant to the assessment of cumulative impacts (see Chapter 5) that could occur at SRS.

Savannah River Site Waste Management Final Environmental Impact Statement

DOE issued the SRS Waste Management EIS (DOE 1995c) to provide a basis for the selection of a sitewide approach to managing present and future (through 2024) wastes generated at SRS. These wastes would come from ongoing operations and potential actions, new missions, environmental restoration, and decontamination and decommissioning programs.

The SRS Waste Management EIS includes the treatment of wastewater discharges in the Efflu-

ent Treatment Facility, F- and H-Area tank operations and waste removal, and construction and operation of a replacement high-level waste evaporator in the H-Area tank farm. In addition, it evaluates the Consolidated Incineration Facility for the treatment of mixed waste. The Record of Decision (60 FR 55249) stated that DOE will configure its waste management system according to the moderate treatment alternative described in the EIS. The SRS Waste Management EIS is relevant to this SNF Management EIS because it evaluates management alternatives for various types of waste that actions proposed in this EIS could generate. The Waste Management EIS is also relevant in the assessment of cumulative impacts that could occur at the SRS (see Chapter 5).

Environmental Impact Statement for a Geologic Repository for the Disposal of SNF and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada

On August 13, 1999, DOE announced the availability (64 FR 44200) of a draft environmental impact statement for a geologic repository at Yucca Mountain for the disposal of SNF and high-level radioactive waste, in accordance with the Nuclear Waste Policy Act of 1982. The DEIS evaluates site-specific environmental impacts from the construction, operation, and closure of the repository. It also evaluates reasonable alternatives for implementing such a proposal, and transportation-related impacts for shipments from across the United States. The DEIS also evaluates the consequences at SRS of continued SNF and high-level waste management assuming the repository is not constructed and operated. The repository decision will affect the ultimate disposal of SNF from SRS. The Final EIS is scheduled to be completed in Fiscal Year 2001.

Treatment and Management of Sodium-Bonded Spent Nuclear Fuel Environmental Impact Statement

DOE has published a draft environmental impact statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

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(64 FR 8553 2/22/99). Alternatives to processing at the Idaho National Engineering and Environmental Laboratory (INEEL) include the use of the Plutonium-Uranium Extraction (PUREX) solvent extraction method currently in use at SRS and the melt and dilute technology that is being proposed under this EIS. The technologies would be applied to sodium-bonded spent nuclear fuel blanket assemblies, which are currently in storage at INEEL. There is approximately 22.4 MTHM of Experimental Breeder Reactor-II (EBR-II) blanket fuel and 34.2 MTHM of Fermi-1 blanket fuel to be processed. This EIS includes cumulative impacts of sodium-bonded spent nuclear fuel processing at the SRS based on estimates from conventional processing of Fuel Group A. Fuel Group A is mostly EBR-II fuel (16.7 MTHM out of 19 MTHM) and therefore provides a good basis for estimating impacts from processing of similar material at SRS. DOE estimates that the impacts for conventional processing would be sufficiently representative of impacts from melt and dilute for the purpose of presenting cumulative impacts.

Management of Certain Plutonium Residues and Scrub Alloy at the Rocky Flats Environmental Technology Site Final Environmental Impact Statement

In August 1998, the Department issued the Final EIS (DOE 1998a). In this EIS DOE proposed to process certain plutonium-bearing materials being stored at the Rocky Flats Environmental Technology Site (Rocky Flats) located near Golden, Colorado. These materials are plutonium residues and scrub alloy remaining from nuclear weapons manufacturing operations formerly conducted by DOE at that site. In their present forms, these materials cannot be disposed of or otherwise dispositioned because they contain plutonium in concentrations exceeding DOE safeguards termination requirements.

DOE has decided to ship approximately 7,450 pounds of sand, slag and crucible and plutonium fluoride residues (containing approximately 600 pounds of plutonium) and ap-

proximately 1,543 pounds of scrub alloy (containing approximately 440 pounds of plutonium) to SRS where these materials will be stabilized in F Canyon by chemically separating the plutonium from the remaining materials in the residues and scrub alloy. The separated plutonium will be placed in safe and secure storage, along with a larger quantity of plutonium already in storage at the Savannah River Site, until DOE has completed the *Surplus Plutonium Disposition Environmental Impact Statement* and made final decisions on the disposition of the separated plutonium. Transuranic wastes generated during the chemical separations will be sent to the Waste Isolation Pilot Plant for disposal. Other wastes generated during the chemical separations operations will be disposed of in accordance with the Savannah River Site's normal procedures for disposing of such wastes. The actions will occur between 1998 and 2002.

Final Environmental Impact Statement Accelerator Production of Tritium at Savannah River Site (DOE, 1998b)

DOE has proposed an accelerator design (using helium-3 target blanket material) and an alternate accelerator design (using lithium-6 target blanket material). If an accelerator is built, it would be located at SRS. In the Record of Decision DOE decided to use an existing commercial light-water reactor as the new tritium source. Therefore, the accelerator will not be built at SRS and impacts from construction and operation are not included in the cumulative impacts section of this EIS.

Final Environmental Impact Statement for the Construction and Operation of a Tritium Extraction Facility at the Savannah River Site (DOE 1998c)

As stated in the Record of Decision (64 FR 26369; 5/14/99), DOE will construct and operate a Tritium Extraction Facility on SRS to provide the capability to extract tritium from commercial light water reactor targets and targets of similar design. The purpose of the proposed action and alternatives evaluated in the

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EIS is to provide tritium extraction capability to support either accelerator or reactor production. The Tritium Extraction Facility EIS is relevant in the assessment of cumulative impacts that could occur at SRS (see Chapter 5).

their current storage configuration. The decisions to use processing capabilities have been documented in a number of Records of Decision, including those following the *F-Canyon Plutonium Solutions EIS*, the *Interim Management of Nuclear Materials EIS*, and the *Rocky Flats Plutonium Residues EIS*. These decisions are consistent with DOE's Implementation Plan for Defense Nuclear Facilities Safety Board Recommendation 94-1, wherein the Board recommended that DOE take steps, including use of the processing facilities, to stabilize nuclear materials that represented health and safety risks.

The Processing Needs Assessment evaluated four material categories that could require the canyons for stabilization or disposition: spent nuclear fuel, plutonium-239, uranium, and other special isotopes. The results of the assessment are being reviewed by DOE management to identify needed follow-on actions.

Other materials under consideration for processing as SRS canyons include various components currently located at other DOE sites, including Oak Ridge, Rocky Flats, Los Alamos, and Hanford. These materials, which were identified during the Processing Needs Assessment, consist of various plutonium and uranium components. If DOE were to process these materials in the SRS separations facilities, additional NEPA reviews would need to be performed. This material has been considered in the cumulative impacts presented in Chapter 5.

Surplus Plutonium Disposition Final Environmental Impact Statement (DOE 1999)

This EIS analyzes the activities necessary to implement DOE's disposition strategy for surplus plutonium. Following completion of the EIS, SRS was selected (65.FR 1608) as the location for mixed oxide fuel fabrication and plutonium immobilization facilities that would be used for plutonium disposition, and for the plutonium pit (a component of nuclear weapons) disassembly and conversion facility. The projected impacts of these operations are incorporated in Chapter 5 of this EIS.

1.6.2 OTHER RELEVANT DOCUMENTS

In August 1997, DOE chartered the Nuclear Materials Processing Needs Assessment. The purpose of the assessment was to determine which, if any, additional nuclear materials within the Department of Energy complex may require use of the SRS chemical separations facilities (F or H canyon) for stabilization or preparation for disposition prior to canyon decommissioning. Chemical separations operations are occurring at SRS because DOE is using the canyons to stabilize nuclear materials that represent potential health and safety risks in

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References

- DOE (U.S. Department of Energy), 1992, "ACTION: A Decision on Phaseout of Reprocessing at the Savannah River Site (SRS) and the Idaho National Engineering Laboratory (INEL) is Required," memorandum to the Secretary of Energy from Assistant Secretary for Defense Programs, Washington, D.C., April 28.
- DOE (U.S. Department of Energy), 1994, *Final Defense Waste Processing Facility Supplemental Environmental Impact Statement*, DOE/EIS-0082-S, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1995a, *Final Environmental Impact Statement, Interim Management of Nuclear Materials*, DOE/EIS-0220, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1995b, *Final Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement*, DOE/EIS-0203, Idaho Operations Office, Idaho Falls, Idaho.
- DOE (U.S. Department of Energy), 1995c, *Savannah River Site Waste Management Final Environmental Impact Statement*, DOE/EIS-0217, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1995d, *Final Programmatic Environmental Impact Statement for Tritium Supply and Recycling*, DOE/EIS-0161, Office of Reconfiguration, Washington, D.C.
- DOE (U.S. Department of Energy), 1996a, *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel*, DOE/EIS-0218F, Washington, D.C.
- DOE (U.S. Department of Energy), 1996b, *Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement*, DOE/EIS-0240, Office of Fissile Materials Disposition, Washington, D.C.
- DOE (U.S. Department of Energy), 1996c, *Storage and Disposition of Weapons-Usable Fissile Materials Programmatic Environmental Impact Statement*, DOE/EIS-0229, Office of Fissile Materials Disposition, Washington, D.C.
- DOE (U.S. Department of Energy), 1998a, *Final Environmental Impact Statement on Management of Certain Plutonium Residues and Scrub Alloy Stored at the Rocky Flats Environmental Technology Site*, DOE/EIS-0277D, Washington, D.C.
- DOE (U.S. Department of Energy), 1998b, *Final Environmental Impact Statement Accelerator Production of Tritium*, DOE/EIS-0270, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1998c, *Final Environmental Impact Statement for the Construction and Operation of a Tritium Extraction Facility at the Savannah River Site*, DOE/0271, Savannah River Operations Office, Aiken, South Carolina

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- EC | DOE (U.S. Department of Energy), 1999, *Surplus Plutonium Disposition Final Environmental Impact Statement*, DOE/EIS-0283F, Office of Fissile Materials Disposition, Washington, D.C.
- 60 FR 65300 (Volume 60 *Federal Register* page 65300), 1995, "Record of Decision for the Interim Department of Energy, Washington, D.C., pp. 65300-65316, December 19.
- 60 FR 28679 (Volume 60 *Federal Register* page 28679), 1995, "Record of Decision for the Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement," Volume 60, Number 105, U.S. Department of Energy, Washington, D.C., pp. 28679-28696, June 1.
- 60 FR 9441 (Volume 60 *Federal Register* page 9441), 1995, "Amendment to Record of Decision for the Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement," Volume 61, Number 74, U.S. Department of Energy, Washington D.C., pp. 9441-9443, March 8.
- 60 FR 18589 (Volume 60 *Federal Register* page 18589), 1995, "Record of Decision for the Defense Waste Processing Facility Supplemental Environmental Impact Statement," Volume 60, Number 70, U.S. Department of Energy, Washington, D.C., pp. 28589-18594, May 12.
- 60 FR 55249 (Volume 60 *Federal Register* page 55249), 1995, "Record of Decision for the Savannah River Site Waste Management Final Environmental Impact Statement," Volume 60, Number 209, U.S. Department of Energy, Washington, D.C., pp. 55249-55259, October 30.
- 60 FR 63877 (Volume 60 *Federal Register* page 63877), 1995, "Record of Decision for the Tritium Supply and Recycling Final Programmatic Environmental Impact Statement," Volume 60, Number 238, U.S. Department of Energy, Washington, D.C., pp. 63877-63891, December 12.
- 60 FR 40164 (Volume 60 *Federal Register* page 40164), 1995, "Notice of Intent to prepare an Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada," Volume 60, Number 151, U.S. Department of Energy, Washington, D.C., pp. 40164-40170, August 7.
- 61 FR 40619 (Volume 61 *Federal Register* page 40619), 1996, "Record of Decision for the Disposition of Highly Enriched Uranium Final Environmental Impact Statement," Volume 61, Number 151, U.S. Department of Energy, Washington, D.C., pp. 40619-40629, August 5.
- 61 FR 25091 (Volume 61 *Federal Register* page 25091), 1996, "Record of Decision for the Final Environmental Impact Statement for Proposed Nuclear Weapons Non Proliferation Policy Concerns for Foreign Research Reactor Spent Nuclear Fuel, DOE/EIS-218F," Volume 61, Number 97, U.S. Department of Energy, Washington D.C., pp. 25091-25103, March 17.
- 61 FR 6633 (Volume 61 *Federal Register* page 6633), 1996, "Supplement Record of Decision for Interim Management of Spent Nuclear Fuel at the Savannah River Site," Volume 61, Number 35, U.S. Department of Energy, Washington D.C., pp. 6633-6637, February 21.

- 61 FR 48474 (Volume 61 *Federal Register* page 48474), 1996, "Supplement Record of Decision for Interim Management of Spent Nuclear Fuel at the Savannah River Site," Volume 61, Number 79, U.S. Department of Energy, Washington, D.C., pp. 48474-48479, September 13.
- 62 FR 17790 (Volume 62 *Federal Register* page 17790), 1997, "Supplement Record of Decision for Interim Management of Spent Nuclear Fuel at the Savannah River Site," Volume 61, Number 70, U.S. Department of Energy, Washington, D.C., pp. 17790-17794, April 11.
- 62 FR 3014 (Volume 62 *Federal Register* page 3014), 1997, "Record of Decision for the Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement," Volume 62, pp. 3014-3030, January 21.
- 62 FR 28009 (Volume 62 *Federal Register* 28009), 1997, "Notice of Intent for the Surplus Plutonium Disposition Final Environmental Impact Statement," Volume 62, Number 99, U.S. Department of Energy, Washington, D.C., pp. 28009-28014, May 22.
- TC | 64 FR 44200 (Volume 64 *Federal Register* 44200), 1999, "Notice of Availability, Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactivity at Yucca Mountain, Nye County, NV," Volume 64, Number 156, U.S. Department of Energy, Washington, D.C., pp. 44200-44202, August 13.
- 64 FR 8553 (Volume 64 *Federal Register* 8553), 1999, "Notice of Availability of the Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel," Volume 64, Number 146, U.S. Department of Energy, Washington, D.C., pp. 41404-41405, July 30, 1999.
- EC | 65 FR 1608 (Volume 65 *Federal Register* 1608), 2000, "Record of Decision for the Surplus Plutonium Disposition Final Environmental Impact Statement," Volume 65, Number 7, U.S. Department of Energy, Washington, D.C., pp. 1608-1620.

CHAPTER 2. PROPOSED ACTION AND ALTERNATIVES

This chapter describes the U.S. Department of Energy's (DOE) proposed action; that is, the management of spent nuclear fuel (SNF) at the Savannah River Site (SRS). Technical terms are defined in the Glossary.

2.1 Proposed Action

As described in Chapter 1, SRS will receive aluminum-based SNF from foreign research reactors, domestic research reactors, and other DOE sites. DOE will have to manage this fuel, in addition to some SNF already stored at the Site, in a manner that will protect human health and the environment. Additionally, DOE is committed to avoiding indefinite storage at SRS of SNF that is in a form unsuitable for final disposition. Therefore, DOE's proposed action is to safely manage SNF that is currently located or expected to be received at SRS, including treating or packaging aluminum-based SNF for possible offsite shipment and disposal in a geologic repository, and packaging non-aluminum clad fuel for on-site dry storage or offsite shipment.

In the Record of Decision (ROD) for the Final Environmental Impact Statement on a Proposed Nuclear Nonproliferation Policy Concerning Foreign Research Reactor SNF (61 FR 25092), DOE stated that it would embark on an accelerated program at SRS to identify, develop, and demonstrate one or more non-chemical processing, cost effective treatment or packaging technologies to prepare aluminum-based foreign research reactor spent nuclear fuel for ultimate disposition.

Based on that decision, DOE's proposal is to select a new non-chemical processing technology that would put aluminum-based foreign research reactor SNF into a form or container suitable for direct placement in a geologic repository. Treatment or conditioning of the fuel would address potential repository acceptance criteria and potential safety concerns. Implementing the new non-chemical processing

treatment or packaging technology would allow DOE to manage the SNF in a road-ready condition at SRS in dry storage pending shipment offsite.

Because of the similarity of the material, DOE proposes to manage the other aluminum-alloy SNF that is the subject of this EIS (domestic research reactor and DOE reactor fuels) in the same manner as the foreign research reactor fuels.

In the Final Environmental Impact Statement on a Proposed Nuclear Nonproliferation Policy Concerning Foreign Research Reactor SNF Record of Decision, DOE stated that, should it become apparent by the year 2000 that DOE will not be ready to implement a new SNF treatment technology, DOE would consider chemically processing foreign research reactor SNF in F Canyon. The Final Environmental Impact Statement on a Proposed Nuclear Nonproliferation Policy Concerning Foreign Research Reactor SNF Record of Decision described the possible use of F Canyon for SNF processing based on a preliminary concept to consolidate all processing operations in one canyon. Subsequent review has shown that consolidating highly enriched uranium spent fuel processing operations in F Canyon would not be practical due to criticality considerations and process capacity restrictions associated with the plutonium-uranium extraction system used in F Canyon. Thus, DOE is now proposing to use H Canyon to chemically separate highly enriched uranium spent fuel.

DOE also committed that any decision to use conventional chemical processing would consider the results of a study (62 FR 20001) on the nonproliferation, cost, and timing issues associated with chemically processing the fuel. DOE stated that any highly enriched uranium separated during chemical processing would be blended down to low enriched uranium.

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EC | DOE has included chemical processing as a management alternative in this EIS, although DOE's preference is to use non-chemical operations processes. DOE proposes to use conventional processing to stabilize some materials before a new treatment facility is in place. The rationale for this is to avoid the possibility of urgent future actions, including expensive recovery actions that would entail unnecessary radiation exposure to workers, and in one case, to manage a unique waste form (i.e., core filter block).

EC | The limited proposed canyon processing actions is not expected to extend the operating schedules for these facilities beyond the current planning basis. Processing would eliminate potential health and safety vulnerabilities that could occur prior to the availability of a new SNF treatment technology. In the event a new treatment process becomes available, the SNF with potential health and safety vulnerabilities could be processed using the new treatment technology.

EC | Previous DOE management decisions on disposition of SNF are outlined in Section 1.1 and Appendix C, Section C.1.2. Relevant National Environmental Policy Act documents are discussed in Section 1.6.

EC | **2.2 Spent Nuclear Fuel Management Technology Options**

TC | DOE has identified 11 potential treatment and packaging technology options in addition to conventional processing that could be used to prepare aluminum-based SNF at SRS for final disposition in a geologic repository. All of the technology options are discussed in Appendix A of this EIS.

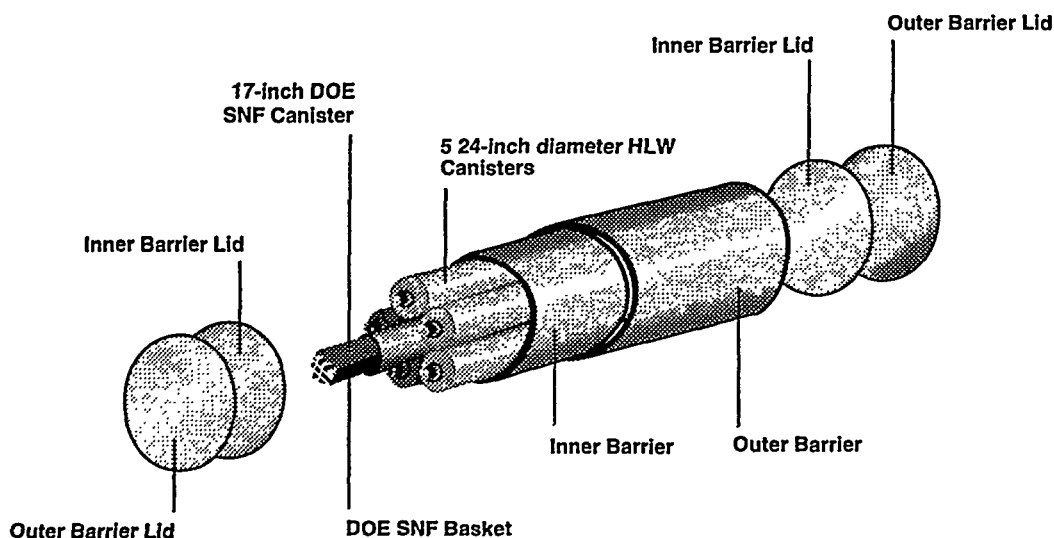
Two of the options, Direct Disposal and Direct Co-Disposal, are non-destructive methods to

prepare and package aluminum-based SNF for disposition in a geologic repository. Another technology option, Repackage and Prepare to Ship, is pertinent only to non-aluminum-clad SNF and programmatic material that would be shipped offsite. These three technology options are discussed under the New Packaging Technology options section (Section 2.2.3) of this EIS.

Nine of the technology options are potential processes for the treatment of aluminum-based SNF. These are Melt and Dilute, Press and Dilute, Chop and Dilute, Plasma Arc Treatment, Glass Material Oxidation and Dissolution System, Dissolve and Vitrify, Electrometallurgical Treatment, Can-in-Canister, and Chloride Volatility. DOE has consolidated seven of these processing technology options into four categories for analysis in this EIS. The Press and Dilute and the Chop and Dilute options are similar, so DOE has represented them for analysis as Mechanical Dilution. The Plasma Arc Treatment, the Glass Material Oxidation and Dissolution System, and the Dissolve and Vitrify options use processes that produce a product with properties similar to that produced at the Defense Waste Processing Facility (DWPF) at SRS. Therefore, DOE has represented these three as the Vitrification option. The Melt and Dilute and the Electrometallurgical Treatment options are analyzed separately. The new treatment options are discussed under the New Processing Technology section of this EIS (Section 2.2.4).

DOE considered the remaining two technology options but dismissed them from analysis in this EIS. With Chloride Volatility, SNF would react with chlorine gas at high temperatures to form volatile chlorides. The uranium, aluminum, fission products, and transuranics would be separated from each other by cooling and distillation. This technology is very immature

Co-Disposal packaging – strategy for all options requiring shipment to the geologic repository



Two alternatives, New Packaging Technology and New Processing Technology, would result in the dry storage of SNF in a road-ready condition. Under these alternatives, the fuel would be contained in stainless-steel canisters. At the repository the canisters would be loaded into a repository waste package with canisters of vitrified high-level waste. DOE expects five canisters of high-level waste would fit in a repository waste package, leaving room for one canister of SNF. This approach is termed Co-Disposal. It would enable repository disposal of SNF with no space requirements beyond those needed for the disposal of the high-level vitrified waste. The high radiation field of the vitrified high-level waste would provide safeguards protection against unauthorized diversion of the SNF for recovery of the enriched uranium.

The SNF canisters would be packaged for Co-Disposal with high-level vitrified waste canisters at the repository,

in terms of actual development and testing and the potential for implementation in a timely manner is very uncertain. In addition, this method of chemical separation offers no advantage over conventional processing and DOE eliminated the option from further consideration.

The second technology option dismissed from analysis was Can-in-Canister, under which DOE would place SNF in a can (in an amount that would not pose criticality concerns), place the can in a stainless-steel canister, and fill the canister with vitrified high-level waste. This technology was originally developed as a means for disposing of immobilized plutonium. Because plutonium does not emit intense penetrating radiation, the high radiation field of the vitrified high-level waste would render the plutonium

inaccessible. However, a more cost-effective and technologically viable way to protect the SNF with radiation fields is to employ the co-disposal concept. Should the Can-in-Canister method be used with aluminum SNF, the high temperature of the molten glass could melt the aluminum in the fuel, changing the geometry of the fuel matrix in an uncontrolled fashion. Therefore, this option could pose significant risks to human health and the environment, and for that reason was not considered a reasonable alternative.

The New Packaging Technology options and the New Processing Technology options consist of several technology options that DOE has not previously applied to the management of aluminum-based SNF for the purpose of ultimate disposition. As a result, DOE believes that the

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highest confidence of success and greatest technical suitability lies with options that have relatively simple approaches (i.e., Direct Disposal, Direct Co-Disposal, Melt and Dilute, and Press and Dilute).

2.2.1 REPOSITORY CONSIDERATIONS

As discussed in Section 2.1, part of DOE's proposed action is to prepare SNF to meet the requirements that the Department anticipates will be applicable to material to be placed in a geologic repository. Any technology that DOE implements must be able to provide a product that is compatible with such criteria. DOE must rely on reasonable assumptions about what the acceptance criteria would include when making decisions on SNF treatment technologies. As described in Chapter 1, DOE anticipates that eventually it will place its aluminum-based SNF inventory after treatment or repackaging in a geologic repository.

As the operator of any geologic repository for SNF, DOE would be responsible for developing acceptance criteria for the material that would be placed in the repository. However, the U.S. Nuclear Regulatory Commission (NRC) would be responsible for licensing the repository. Therefore, DOE is working closely with the NRC to develop acceptance criteria. DOE will provide the NRC with characterization data for material that would be prepared for disposal in a geologic repository. At this time, acceptance criteria need to be conservative because of uncertainties concerning any engineered or natural barriers at a repository. However, as repository and packaging designs evolve, the criteria will become more detailed. Fuel characterization data will need to be detailed enough to verify that each element or canister falls within the ultimate acceptance criteria. Such detail, however, is not currently available. Final acceptance criteria will not be available until after NRC issues its authorization, based on the successful demonstration of safe, long-term performance of the candidate repository in accordance with NRC regulations. Until such time, the preliminary acceptance criteria tend to be conservative to allow for uncertainties in performance of en-

gineered or natural barriers and how such performance may impact public and worker health and safety, and material isolation.

DOE has performed preliminary evaluations of the expected SNF characteristics (DOE 1995a, 1996a). Those evaluations indicated that the SNF to be placed in the repository would have to meet requirements for the following characteristics:

Packaging

- Dimension and weight limits
- Material compatibility
- Thermal limits
- Internal gas pressure limits
- Labeling
- Handling ability
- Waste isolation

Contents

- Solid material – no particulates
- Noncombustible
- No free liquids
- No hazardous waste (as defined by the Resource Conservation and Recovery Act)

Chemical reactivity

- Not chemically reactive
- Nonpyrophoric
- Nonexplosive

Nuclear material safeguards

- Reduced uranium-235 enrichment
- Self-protecting radiation fields
- Tamper-proof seals

Criticality control

- Limits on nuclear reactivity by controlling amount of uranium and its enrichment (see Text Box on page 2-5)

Proliferation and Criticality Concerns for SNF Disposal

Preparation of SNF for disposal in a geologic repository requires consideration of the risk of a disruptive nuclear criticality. Criticality risk is defined as the potential for a neutron-induced self-sustaining fission reaction like that which occurs in a nuclear reactor. Nuclear criticality in the SNF would be due to uranium enriched in the fissile nuclide uranium-235 with the remainder being principally non-fissile uranium-238. Characteristic enrichment levels in these fuels are designated as follows (DOE 1996b).

	Percent uranium-235
Highly enriched uranium (HEU)	>20-93
Low enriched uranium (LEU)	>2-≤20
Commercial power reactor fuel	<2-4
Very low enriched uranium (VLEU)	≤2
Natural uranium (NU)	0.72
Depleted uranium (DU)	Typically 0.18

Concern for the enrichment level of the fuel arises from two considerations: (1) weapons material proliferation policy and (2) criticality control during storage, transportation, and repository disposal. The high-enriched uranium fuels are generally considered to present unacceptable proliferation risks, unless otherwise protected. Isotopic dilution of the high-enriched uranium fuels to 20 percent uranium-235 during treatment for repository disposal satisfies requirements for protection against this proliferation risk.

One approach to control the potential for a nuclear reaction during storage, transport, and repository disposal of the SNF (high-enriched uranium or low-enriched uranium) is addressed by incorporation of neutron-absorbing poison materials in the waste form or containers, by reduction of enrichment levels to the extent practical (2 to 20 percent), and by limiting the mass loading of fissile uranium-235 in the primary waste form canisters. Provisional limits for fissile mass loadings have been specified as follows (DOE 1996b):

	Allowable fissile mass loading (kg U-235) per canister*
HEU	14.4
LEU	43
VLEU	200

*Larger quantities of fissile U-235 in the canister are permitted at lower enrichment levels because of neutron escape or absorption in non-fissile material.

In accord with these specifications, the SNF processed for Direct Co-Disposal (with no dilution of highly enriched uranium) would require incorporation of neutron poisons in the waste canister and possibly smaller canisters to meet fissile mass loading limits. The processes under the New Processing Technology, which would achieve enrichment levels of 20 percent or less, would generate canisters within the low-enriched uranium fissile mass loading limits, but could require incorporation of poison materials for additional criticality protection.

Radiation

- Radiation field limits
- Canister surface contamination limits

The preliminary acceptance criteria describe the physical, chemical, and thermal characteristics to which spent nuclear fuel, high-level waste, and associated disposable canisters must con-

form for emplacement in the repository. The preliminary criteria are organized into four categories:

- General/Descriptive
- Physical/Dimensional
- Chemical/Compatibility
- Thermal/Radiation/Pressure

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Disposability Assessment: Aluminum-Based Spent Nuclear Fuel Forms (WSRC 1998a) provides a technical assessment of the Melt-and-Dilute and Prepare for Direct Disposal/Direct Co-Disposal technologies against these preliminary criteria. This assessment is based on results of several analytical and experimental – investigations at SRS, and criticality calculations. The Disposability Assessment concluded:

Both Melt-Dilute and Direct [disposal] forms [for aluminum-alloy SNF] in disposable containers can meet the requirements of the Draft Standards for Spent Nuclear Fuel in Disposable Canisters. Completed analyses indicate that the Melt-Dilute form of eutectic composition (13.2 percent [uranium]) and containing less than 20 percent ²³⁵U [uranium-235] meets the requirements of the draft standards. Additional criticality analysis of the Melt-Dilute form and HLW [high-level waste] degraded within a waste package are needed for the disposability assessment and are being scheduled for FY00 and subsequent years as part of the development process for the full scale facility. The Melt-Dilute form is flexible in that additional dilution or the addition of neutron poisons to the Melt-Dilute product can be readily made, if necessary.

The Direct form in disposable canisters can meet all requirements of the Draft Standards. Criticality analyses have identified that neutron poison additions are needed to preclude criticality of degraded Al-SNF [aluminum based spent nuclear fuel] within a canister and of degraded Al-SNF and HLW within a waste package. A method is needed to incorporate neutron poisons into the canisters in the demonstration that reactivity of all possible configurations is within the acceptable limit. Several poison materials have been suggested and are being evaluated and tested for compatibility with the Al-SNF. These activities will continue throughout the

development process for the full scale melt and dilute facility.

Based on the preliminary criteria and the conclusions in the Disposability Assessment, preliminary judgments can be made regarding the acceptability for disposal of the final waste forms produced under the other technologies evaluated in this EIS. Final disposal requirements will be specified by NRC; currently the final waste form produced under the Conventional Processing technology (borosilicate glass) is the best demonstrated available technology for treatment of high-level waste (55 FR 22520). Therefore, DOE has high confidence that this waste form would be acceptable for disposal in a geologic repository. The final waste form produced under the Vitrification technologies and Electrometallurgical Treatment technologies is similar to that produced under the Conventional Processing technology; thus, DOE also would have high confidence in the acceptability of their final products. For Vitrification technologies, criticality and nonproliferation concerns would need to be addressed by the dilution of the highly-enriched uranium to low-enriched uranium.

The solid form with low enrichment that would be the product of mechanical dilution could be acceptable for storage in a geologic repository. However, this technology would not be as effective from a nuclear nonproliferation perspective as other treatments (such as Melt and Dilute) because of the potential to separate the pressed or chopped depleted uranium and SNF.

Nuclear materials safeguards are one of the most important issues to be addressed for both onsite storage and transportation to a repository. Much of the aluminum-based SNF contains appreciable quantities of highly enriched uranium or plutonium. In addition to secure management, there are two basic methods for ensuring that these fissile materials have the proper safeguards: (1) reducing the uranium-235 enrichment or (2) making the fuel self-protecting. Reduced uranium-235 enrichment makes the fissile materials incapable of producing a nuclear explosion. Reenrichment would require a

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massive commitment of resources not available to most nations. "Self-protecting" means the radiation fields around the fuel are sufficiently high that recovery of the fissile materials would be impossible without the considerable resources of facilities such as those at SRS.

Finally, the integrity of the fuel form that is stored after treatment pending shipment to a repository must be sufficient to ensure safe interim storage and to prevent degradation of design features that may be relied upon in the repository.

TC | Because the melt and dilute waste form could eventually be disposed of in a geologic repository, DOE-SR signed in August 1997 a Memorandum of Understanding with the NRC for its review of the research effort that DOE-SR is conducting. DOE-SR has provided the NRC with several technical reports on the results obtained from the research effort. Based upon its initial review, the NRC in a June 1998 letter (Knapp 1998) stated that "both the direct co-disposal and melt-dilute options would be acceptable concepts for the disposal of aluminum-based research reactor SNF in the repository." Additionally, as research efforts yield new findings, DOE is providing the information to the NRC.

TC | DOE would not implement a treatment technology option unless it has a high degree of confidence that the technology option would produce a final form that was compatible with what DOE believes the repository acceptance criteria will be. In order to ensure that the treatment technology DOE could select will produce a product that is likely to meet the acceptance criteria, DOE-SR is working with the NRC to obtain comments on the research and development work that DOE will perform to establish treatment technology specifications. To provide additional confidence in the suitability of new treatment technologies, DOE requested that the National Academy of Sciences (NAS) evaluate and provide recommendations regarding DOE's aluminum-based SNF disposition technical development program. Results of the NAS review are summarized in Section 2.6.1.

2.2.2 FACILITIES

Under the alternatives considered in this EIS, the Department could need a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility. A Transfer and Storage Facility for SNF would provide remote handling and heavy lifting capability, hot cells, and space to receive SNF shipments; place the SNF in interim storage as needed; open the shipping containers; sample and analyze the fuel; crop end fittings if necessary; vacuum-dry the SNF; repackage the fuel into storage canisters; and place the repackaged fuel in dry interim storage. Section 2.3.2.1 provides information on the Transfer and Storage Facility. A Transfer, Storage, and Treatment Facility would provide the capability to implement the options of the New Processing Technology. Section 2.3.2.2 provides more information on the Transfer, Storage, and Treatment Facility.

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For all technologies, DOE would continue to use the Receiving Basin for Offsite Fuel and the L-Reactor Disassembly Basin for currently stored SNF and to receive and store incoming fuel. If DOE built the Transfer and Storage Facility, newly received fuel could go to that facility, and the inventory in the wet basins would gradually be moved to new dry storage. DOE intends to discontinue wet storage by 2009 (DOE could continue to use the L-Reactor Disassembly Basin for SNF receipt and unloading if Building 105-L was modified as a Transfer, Storage, and Treatment Facility [see Section 2.3.2]).

All currently stored SNF at the SRS is located in the Receiving Basin for Offsite Fuel or the L-Reactor Disassembly Basin (generically termed "wet basins" in this EIS). DOE initially would receive and store incoming fuel either in the L-Reactor Disassembly Basin or the Receiving Basin for Offsite Fuel and begin construction of a new Transfer and Storage or Transfer, Storage, and Treatment Facility. Fuel would be transported from wet storage basins to the new facility as prescribed to prepare the material for disposition. Radiological consequences of the on-site transportation of the spent nuclear fuel,

TC | under both incident-free and accident conditions are projected in Section 4.1.1.7.

EC | **2.2.3 NEW PACKAGING TECHNOLOGY OPTIONS**

TC | In this section DOE describes technology options (Direct Disposal/Direct Co-Disposal) that could be used to prepare aluminum-based SNF for placement in a geologic repository and a technology option (Repackaging and Prepare to Ship) that DOE could use to transfer non-aluminum-clad SNF and programmatic material to dry storage pending offsite shipment.

The Direct Disposal/Direct Co-Disposal technology has the advantage of being one of the simplest to implement because it would not require a Treatment Facility, nor would it entail many operational activities. However, several potential technical issues associated with the repository must be resolved. The acceptability of aluminum-based, highly-enriched uranium fuel in a geologic repository is uncertain because of criticality concerns. DOE proposes to address this matter by limiting the amount of uranium permitted in a canister of fuel and by adding a neutron poison. Hydrogen could be produced from radiolysis of bound water in the aluminum metal fuel; however, DOE could minimize hydrogen production by adequate drying and venting, if necessary. The level of SNF characterization and certification requirements is uncertain. DOE expects the operational history of the fuel and some statistical analysis, combined with an evaluation of the more important chemical and physical characteristics (e.g., original fissile material loading, post irradiation burn-up and radiation levels) should be sufficient to characterize the fuel. The need for more detailed characterization information, based on regulatory requirements that will be developed in the future, could require much more costly and time-consuming analysis for each fuel.

2.2.3.1 Prepare for Direct Disposal/Direct Co-Disposal

EC | In the Transfer and Storage Facility, the SNF would be cropped (cropping removes the end pieces of the assembly; see Glossary), vacuum dried, and placed in a stainless-steel canister with a neutron poison. The canisters would be filled with an inert gas, welded closed, and placed in dry storage to await shipment to the geologic repository. Some of the uranium oxide and uranium silicide fuels could require cutting or other resizing to fit into the canisters. As an alternative, special packaging could be used for these oversized fuels.

TC | From an SRS perspective, Direct Disposal and Direct Co-Disposal are identical except for a slight difference in number of canisters produced. The analyses in this EIS would apply equally to either technology. If DOE used canisters with a diameter of about 17 inches (43 centimeters), it could co-dispose (see text box on page 2-3 on the co-disposal concept) the canisters at the repository with vitrified high-level waste prepared in DWPF (Direct Co-Disposal). Otherwise, using 24-inch (61-centimeter) diameter canisters, DOE could dispose of the fuel between waste packages of commercial SNF (Direct Disposal).

Due to the nature and form of the SNF to be managed at SRS, DOE does not expect the Direct Disposal/Direct Co-Disposal technology option would be applicable to all the aluminum-based SNF considered in this EIS. Table 2-1 presents an explanation of the SNF that DOE considers appropriate for the Direct Disposal/Direct Co-Disposal option.

Figure 2-1 shows the Direct Disposal/Direct Co-Disposal option. Appendix A provides a more complete discussion of Direct Disposal and Direct Co-Disposal.

EC | **2.2.3.2 Repackage and Prepare to Ship to Other DOE Sites**

This technology option would apply to two specific fuel groups, and this is the only option considered for these fuel groups.

- DOE has designated management responsibilities for the stainless-steel and zirconium-clad fuels (Group F) to the Idaho National Engineering and Environmental Laboratory (60 FR28680). DOE analyzed the environmental impacts of shipping these non-aluminum-clad fuels to the Idaho National Engineering and Environmental Laboratory in the Programmatic SNF EIS (DOE 1995b).

- TC |
- The Higher Actinide Targets would be stored pending an evaluation of their disposition. Under the Repackaging and Prepare to Ship to Other DOE Sites technology option, DOE evaluates repackaging the Mark-51 and other targets to place them in a new dry storage facility in the event disposition decisions have not been made by the time an SRS dry storage facility is operational.

DOE would not apply the Repackaging and Prepare to Ship option to the Mark-18 targets due to potential health and safety vulnerabilities as described in Section 1.5 of this EIS.

In the Transfer and Storage Facility, the SNF and the Mark-51 and other targets could be cropped, vacuum dried, and placed in stainless-steel canisters, possibly with a neutron poison. The canisters would be filled with an inert gas, welded closed, and placed in dry storage to await shipment off-site. Figure 2-2 shows the Repackage and Prepare to Ship option which would be implemented only in parallel with an alternative that required the construction of a Transfer and Storage Facility or Transfer, Storage, and Treatment Facility. A new facility would not be constructed solely to repackage non-aluminum-based fuels and the higher actinide targets.

EC |

2.2.4 NEW PROCESSING TECHNOLOGY OPTIONS

The New Processing Technology options would reduce the uncertainty associated with placing aluminum SNF in a geologic repository because criticality concerns would be reduced through the opportunity to adjust enrichment, add neutron absorbers, and better control geometry.

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Under these technology options, DOE initially would receive and store incoming fuel either in the L-Reactor Disassembly Basin or the Receiving Basin for Offsite Fuel. DOE would construct and operate a Transfer, Storage, and Treatment Facility (Section 2.3.2.2) to receive later shipments, and would begin to transfer the fuel inventories in the existing storage pools to this facility. DOE could use the dry storage capacity of the facility to store SNF awaiting processing and to store the processed fuel form in a road-ready condition awaiting shipment to the geologic repository.

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If a new facility was built, DOE would phaseout operation of the L-Reactor Disassembly Basin and the Receiving Basin for Offsite Fuel by 2009. In the event that Building 105-L was modified to function as the Transfer, Storage, and Treatment Facility, SNF would continue to be received and unloaded in the L-Reactor Disassembly Basin, but long-term SNF storage in the basin and in the Receiving Basin for Offsite Fuel would be phased out. The Transfer, Storage, and Treatment Facility could be located in a new or existing facility in one of the reactor areas or in a new facility in F or H Area.

Each technology option that DOE could use in the Transfer, Storage, and Treatment Facility, except Electrometallurgical Treatment, would result in an SNF form that DOE would store in road-ready condition. The use of 17-inch (43-centimeter) diameter canisters would support the co-disposal concept; however, DOE could use other canister sizes. DOE assumed a 17-inch canister for purposes of estimating costs of each technology (see Section 2.6.5). The analyses in this EIS would apply equally to other canister sizes.

Table 2-1. Applicability commentary of the New Packaging Technology options.

Fuel group		Prepare for Direct Disposal/Direct Co-Disposal	Repackage and Prepare to Ship to Other DOE Sites
EC	A. Uranium and Thorium Metal Fuels	Applies - These reactive metal fuels would require rigorous drying (hot vacuum drying) to ensure dehydrating and passivation of uranium metal for both short-term and repository storage.	Does not apply - The Record of Decision (60 FR 28680) for the Programmatic SNF EIS (DOE 1995b) determined that DOE would manage aluminum SNF at SRS. DOE would not ship aluminum-based SNF to another site for storage.
	B. Materials Test Reactor-Like Fuels	Applies - The fissile mass loading of the canisters would be limited because of criticality concerns. DOE and NRC ^a are discussing packaging restrictions which would eliminate the possibility of criticality.	Does not apply - The Record of Decision (60 FR 28680) for the Programmatic SNF EIS (DOE 1995b) determined that DOE would manage aluminum SNF at SRS. DOE would not ship aluminum-clad SNF to another site for storage.
	C. HEU/LEU ^b Oxides and Silicides Requiring Resizing	Applies - These fuels would not fit into the 17-inch (43-centimeter) diameter canister without resizing or special packaging. The highly enriched fuels present criticality concerns. The fissile mass loading of the canisters would be limited.	Does not apply - The Record of Decision (60 FR 28680) for the Programmatic SNF EIS (DOE 1995b) determined that DOE would manage aluminum SNF at SRS.
EC	D. Loose Uranium Oxide in Cans	Does not apply - Group D fuels are granular and might contain particulates. Current understanding of acceptance criteria for the geologic repository would rule out acceptance of particulate fuels.	Does not apply - The Record of Decision (60 FR 28680) for the Programmatic SNF EIS (DOE 1995b) determined that DOE would manage aluminum SNF at SRS and would ship non-aluminum fuel to INEEL.
	E. Higher Actinide Targets	Does not apply - This fuel group will be continually wet stored until DOE decides on their final disposition.	Applies - In the future, DOE might decide to ship these targets to another DOE site. Application of this technology to Group E fuels would include only the preparation for shipment, not the shipment itself.
	F. Non-Aluminum-Clad Fuels	Does not apply - The Record of Decision for the Programmatic SNF EIS designated INEEL ^c as the location for management of non-aluminum-clad SNF. SRS activities for Group F fuels are to prepare it for shipment to INEEL.	Applies - Under the Record of Decision (60 FR 28680) for the Programmatic SNF EIS (DOE 1995b), DOE would ship non-aluminum-clad spent nuclear fuel to INEEL. DOE analyzed shipment from wet basins (DOE 1995b) which could occur under the No-Action Alternative. This technology would provide an additional action of repackaging and dry-storing Group F fuel before shipment.

a. NRC = U.S. Nuclear Regulatory Commission.

b. HEU/LEU = Highly Enriched Uranium/Low Enriched Uranium.

c. INEEL = Idaho National Engineering and Environmental Laboratory.

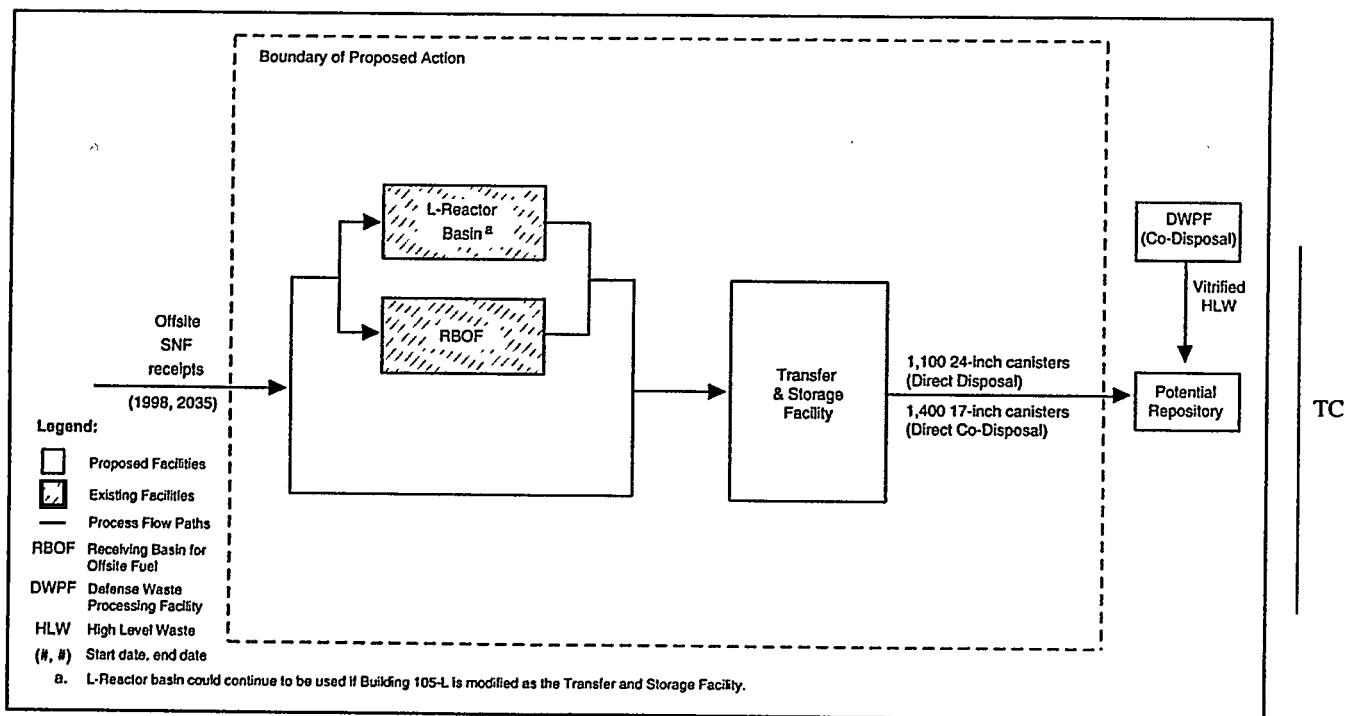


Figure 2-1. New Packaging Technology – Direct Disposal/Direct Co-Disposal.

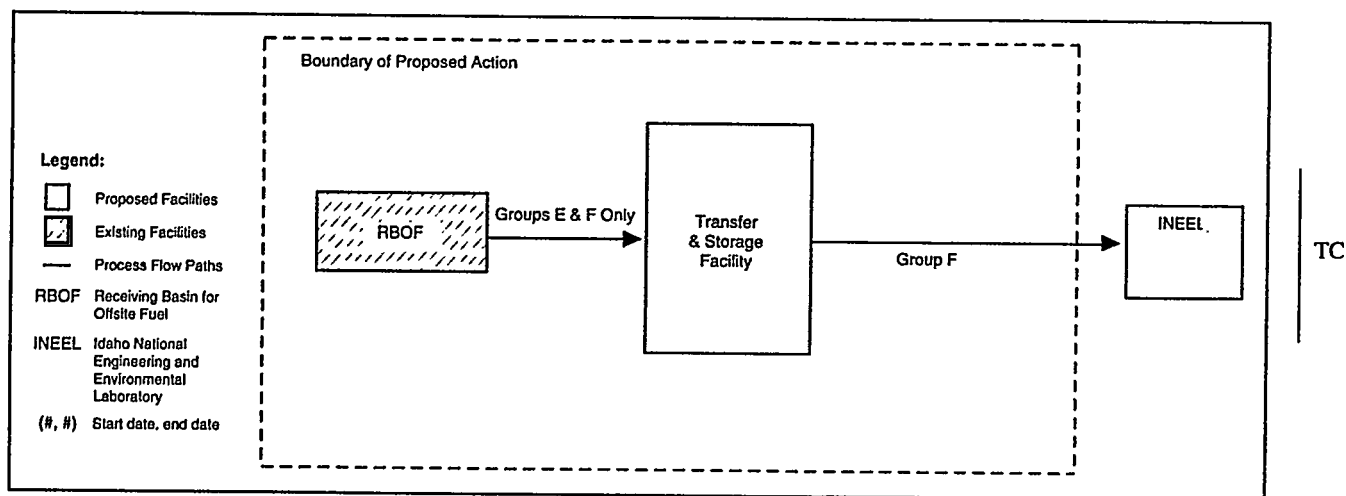


Figure 2-2. New Packaging Technology – Repackage and Prepare to Ship to Another DOE site.

Figures 2-3 and 2-4 show the New Processing Technology options. The following sections describe the new technology options; Appendix A describes them in more detail. Table 2-2 lists the applicability of the New Processing Technology to the fuel groups described in Chapter 1.

2.2.4.1 Melt and Dilute

Under the Melt and Dilute option, DOE would receive, unload, and crop the SNF in the Transfer, Storage, and Treatment Facility and either package the fuel in canisters for placement in dry storage pending treatment or send it directly to the treatment phase. The SNF would be melted and, if highly enriched, mixed with de-

pleted uranium and additional aluminum as necessary to produce a low-enriched uranium-aluminum melt. Neutron poison material also could be added if necessary. The low-enriched uranium product would be placed in corrosion-resistant canisters. The canisters, about 17-inch diameter by 120-inch length (43 by 305 centimeters), would be filled with an inert gas, welded closed, and placed in dry storage to await shipment to the geologic repository.

Under this option, most of the fission products would remain in the uranium-aluminum melt; however, some would be volatilized. Dilution to low enrichment would address nuclear proliferation concerns relating to transport and disposal of fuels. Both the dilution and the poison addition would address criticality concerns. Other characteristics promoting acceptability of the final form for disposal in the geologic repository are discussed in Appendix A.

Based on recent research and development work, preliminary conceptual design work, and considering aspects such as technical maturity, DOE considers Melt and Dilute to be the most viable of the technology options for implementation at SRS. DOE believes Melt and Dilute would entail the least technical risk because DOE has made substantial progress in the development of the melt and dilute process and ongoing work indicates full-scale operations that melt aluminum-based SNF and isotopically dilute the high-enriched uranium are achievable. A review by the National Academy of Sciences indicated that the Melt and Dilute process, as proposed by the SRS, should be achievable for aluminum-based SNF to be managed at SRS.

During the development of the Melt and Dilute technology, DOE may determine that, for technical, regulatory, or cost reasons, the Melt and Dilute option is no longer viable. As a back-up to Melt and Dilute, DOE will continue to pursue the Direct Co-Disposal option of the New Packaging Technology and would attempt to implement this option if Melt and Dilute were no longer feasible or preferable. Direct Co-Disposal has the potential to be the least complicated of the new technology options. How-

ever, there is uncertainty that aluminum-based SNF, packaged according to the Direct Co-Disposal option, would be acceptable in a geologic repository. A comparison of the preferred and backup technologies for aluminum-based nuclear fuel disposal is presented in Table 2-3.

The DOE-SR and the NRC have established an agreement for the NRC to provide technical assistance in connection with the identification of potential issues relating to the placement of aluminum-based foreign and domestic research reactor spent nuclear fuel in a geologic repository. In a review of DOE's research and development work, the NRC staff indicated that both the Melt and Dilute and Direct Co-Disposal technologies would be acceptable concepts for the disposal of aluminum-based research reactor SNF in a repository (Knapp 1998).

2.2.4.2 Mechanical Dilution

For this option, DOE would use a mechanical process to consolidate the fuel and isotopically dilute the uranium-235. The process could be either Press and Dilute or Chop and Dilute (see Appendix A). The impact analyses in Chapter 4 are based on Press and Dilute because DOE believes those impacts would be representative of both technologies, which would have nearly identical process flows, facility requirements, and resulting fuel forms.

DOE would crop and cold-vacuum-dry SNF in the Transfer, Storage, and Treatment Facility and either place the fuel in canisters for dry storage pending treatment or send the fuel directly to the treatment phase for volume reduction and dilution. The Press and Dilute method would flatten fuel assemblies and press them into a laminate between layers of depleted uranium to produce packages with a low overall enrichment. The Chop and Dilute method would shred the fuel and mix it with depleted uranium. Regardless of the dilution method, DOE would package the product in 17- by 120-inch (43- by 305-centimeter) canisters. The package could contain a nuclear poison (in either the laminate or the container) to reduce the

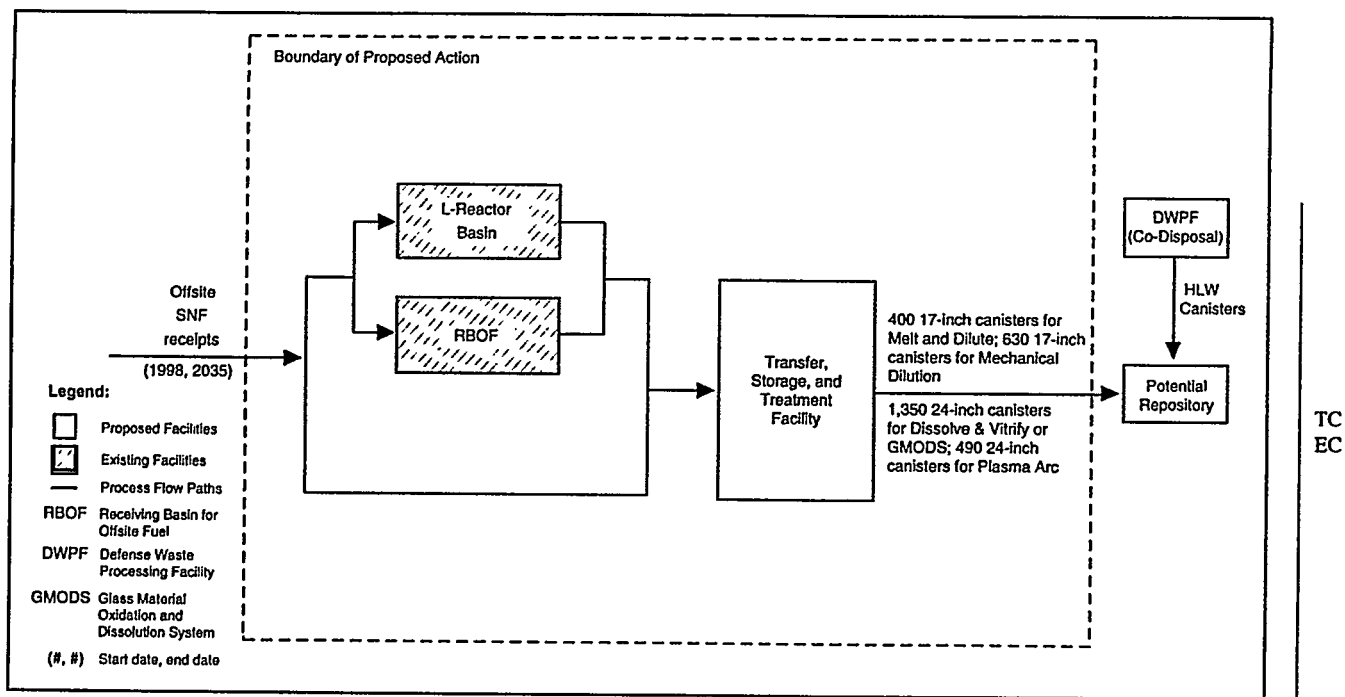


Figure 2-3. New Processing Technology - Melt and Dilute, Mechanical Dilution, Vitrification Technologies.

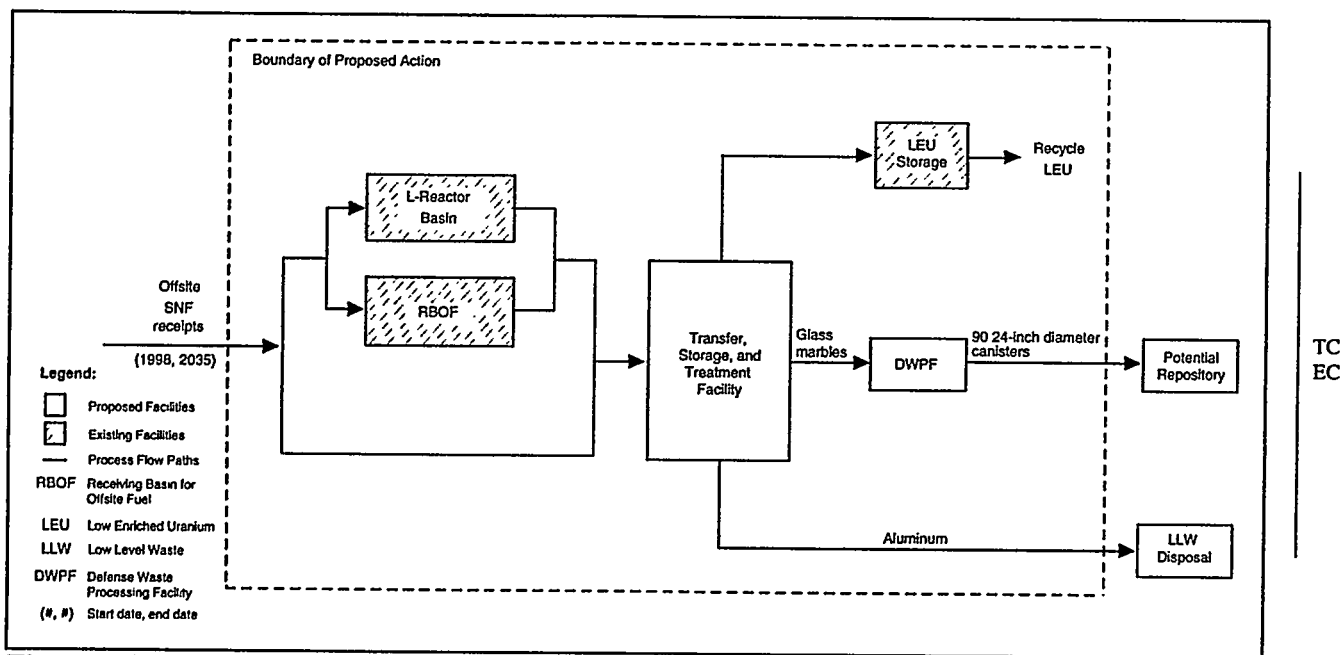


Figure 2-4. New Processing Technology - Electrometallurgical Treatment.

Table 2-2. Applicability of New Processing Technology options.

Fuel Group	Melt and Dilute	Mechanical Dilution	Vitrification Technologies	Electrometallurgical Treatment
A. Uranium and Thorium Metal Fuels	Applies	Does not apply - Mechanical treatment would not address chemical reactivity issue.	Applies	Applies
B. Materials Test Reactor-Like Fuels	Applies	Applies	Applies	Applies
C. HEU/LEU ^a Oxides and Silicides Requiring Resizing	Applies	Applies	Applies	Applies
D. Loose Uranium Oxide in Cans	Applies	Does not apply - These fuels are granular and might contain particulates. This technology would leave Group D fuels as particulates. Current understanding of repository acceptance criteria is that particulate fuels would not be accepted without special treatment.	Applies	Applies
E. Higher Actinide Targets	This fuel group will be continually wet stored until DOE decides on their final disposition.	This fuel group will be continually wet stored until DOE decides on their final disposition.	This fuel group will be continually wet stored until DOE decides on their final disposition.	This fuel group will be continually wet stored until DOE decides on their final disposition.
F. Non-Aluminum-Clad Fuels	Does not apply - Record of Decision for Programmatic SNF EIS ^b designated INEEL ^c as location for non-aluminum SNF management.	Does not apply - Record of Decision for Programmatic SNF EIS designated INEEL as location for non-aluminum SNF management.	Does not apply - Record of Decision for Programmatic SNF EIS designated INEEL as location for non-aluminum SNF management.	Does not apply - Record of Decision for Programmatic SNF EIS designated INEEL as location for non-aluminum SNF management.

a. HEU/LEU = highly enriched uranium/low enriched uranium.

b. DOE (1995b).

c. INEEL = Idaho National Engineering and Environmental Laboratory.

Table 2-3. Comparison of preferred and backup technologies for aluminum-SNF disposal.

Technology	Advantages	Disadvantages
Preferred technology: Melt-Dilute Process	- U-235 enrichment readily adjusted by dilution with depleted uranium to meet proliferation policy and nuclear criticality constraints. - Melting reduces the volume of the fuel (see Section A.2.1). DOE estimates about 400 canisters would be generated, in comparison to about 1,400 canisters for Direct Co-Disposal. - Homogenous melt product provides basis for predictable behavior in geologic repository.	Implementation requires high temperature operation of melter and offgas control equipment in shielded cell.
Backup technology: Direct Co-Disposal Process	- Process technically straightforward to implement. Shielded-cell handling procedures well developed. - Meets non-proliferation policy criteria better than other alternatives.	- Different SNF configurations, materials, and U-235 enrichments present packaging complexities. - No adjustment of U-235 enrichment possible to meet criticality constraints in a geologic repository. May require the use of exotic nuclear poisons. - No reduction in the volume of the fuel. - Non-uniform SNF structures and compositions complicates documentation of fuel characteristics to meet repository waste acceptance criteria and to predict behavior in a geologic repository.

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potential for criticality. The canisters would be filled with an inert gas, welded closed, and placed in dry storage to await shipment to the geologic repository.

The fission products would remain with the uranium-aluminum alloy, making their release difficult. However, mechanical dilution would not be as effective from a nuclear nonproliferation viewpoint as other treatments (such as Melt and Dilute) because of the potential to separate the pressed or chopped depleted uranium and SNF. The dilution process and the addition of a neutron poison would decrease criticality potential. The solid form with low enrichment could be acceptable at the geologic repository. Although hydrogen generation in the canister would be possible due to the radiolysis of bound water, DOE could minimize hydrogen buildup by eliminating water from the canisters (e.g., by vacuum drying).

2.2.4.3 Vitrification Technologies

DOE could use one of three vitrification technologies: (1) Dissolve and Vitrify, (2) Glass Material Oxidation Dissolution System, or (3) Plasma Arc Treatment. In the vitrification options, the SNF would be converted to oxide and dissolved in molten glass to form a vitrified product. These options have the advantage of producing a vitrified waste form similar to that used for the disposal of high-level waste. Therefore, they should qualify for acceptance at a geologic repository. The final form would contain fission products, and criticality and nonproliferation concerns would be addressed by the dilution of enriched uranium.

For these options, DOE would crop and cold-vacuum-dry SNF in the Transfer, Storage, and Treatment Facility and either place the fuel in canisters for dry storage pending treatment or send it immediately for treatment. The resulting glass or ceramic would be poured into 24- by 120-inch (61- by 305-centimeter) canisters and

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placed in dry storage. The use of 24-inch diameter canisters would enable disposal like vitrified high-level waste.

These are advanced technologies. As such, they introduce more technical and schedule risk than the other options in this alternative. This EIS analyzes the impacts of the Dissolve and Vitrify option as representative of all three because DOE believes that the impacts among the three would be similar. The following paragraphs describe the three vitrification technologies; Appendix A provides more information.

Dissolve and Vitrify

The Dissolve and Vitrify treatment is similar to conventional processing except there would be no recovery of enriched uranium. The SNF would be cropped and charged to an electrolytic dissolver. The electrolyte solution would be nitric acid saturated with boric acid. If necessary, depleted uranium would be added to produce low-enriched uranium. The entire solution, including uranium and fission products, would be vitrified. The process would operate in a batch mode to ensure criticality control.

This EIS analyzes performing the Dissolve and Vitrify option in the Transfer, Storage, and Treatment Facility; however, DOE could modify one of the canyons to perform the process. DOE is not considering vitrification of this material in DWPF because that process is not designed to accommodate more than trace quantities of fissile material without major modifications that would be impractical and incompatible with DWPF operations, schedules, and mission.

Glass Material Oxidation and Dissolution System

The Glass Material Oxidation and Dissolution System would convert SNF directly to borosilicate glass using a batch process. The final form would address criticality concerns by diluting the uranium-235 with depleted uranium and by using boron oxide as a dissolving agent (boron is a neutron poison).

The process would use lead dioxide to oxidize the metals in the SNF so they would be soluble in glass. The resulting lead metal would be recovered and oxidized for reuse. The product of the process would be glass marbles that a second stage of melting could consolidate into logs. The process would occur in the new Transfer, Storage, and Treatment Facility.

Plasma Arc Treatment

The Plasma Arc Treatment technology would use a plasma torch to melt and oxidize the SNF in a rotating furnace. The fuel would be fed into the process with minimal sizing or pretreatment. The plasma torch would heat the fuel to temperatures as high as 2,900°F (1,600°C). The rotation of the furnace and the pressure of the torch would mix the melted fuel. A ceramic binder such as contaminated soil would be added to the mixture to form a glass-ceramic. Depleted uranium could be added to the process to produce low-enriched uranium. When the melting and oxidation is complete, the furnace rotation would slow and the molten fuel would flow by gravity into molds. The process would be conducted in the Transfer, Storage, and Treatment Facility, which would be equipped to capture volatile and semivolatile off-gasses.

2.2.4.4 Electrometallurgical Treatment

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Under the Electrometallurgical Treatment option, DOE would crop and cold-vacuum-dry the SNF in the Transfer, Storage, and Treatment Facility, can it, and either place it in dry storage pending treatment or send it immediately to the treatment phase, which would shred and melt it into metal ingots. An ingot would be placed in an electrorefiner, where most of the metal in the SNF (aluminum) would be removed as a low-level waste stream. The remaining metal would be placed in a second electrorefiner where the uranium would be removed. If necessary, the uranium would be fed to a melter where depleted uranium would be added to produce low-enriched uranium. The uranium could be sold as recycled uranium for manufacture into commercial nuclear fuel. The remainder of the fuel materials would be oxidized in a furnace and

dissolved in glass which would be poured into 24- by 120-inch (61- by 305-centimeter) canisters and placed into dry storage.

This option has the advantage of potentially recycling the enriched uranium. Criticality concerns would be addressed by the isotope dilution of the highly enriched uranium, eliminating the issue of SNF acceptance at a geologic repository. DOE has been developing the electrometallurgical treatment process for certain non-aluminum-based SNF.

Figure 2-4 shows the Electrometallurgical Treatment technology. Appendix A provides a more complete discussion of the technology.

2.2.5 CONVENTIONAL PROCESSING TECHNOLOGY

In this technology, DOE would process SNF in the F or H Area Canyon directly from wet storage. The Record of Decision for the Final EIS on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (61 FR 25091) stated that fuel would be processed in F Canyon. Because F Canyon is scheduled to be shut down before all the fuel could be processed, and because F Canyon is not suitable for highly-enriched uranium processing without modifications, H Canyon also would be used. The process would chemically dissolve the fuel and separate fission products from the uranium by solvent extraction. The uranium would be blended with depleted uranium, as necessary, to bring the enrichment down to about 5 percent or less. The wastes from solvent extraction would contain the highly radioactive fission products, thorium, and possibly some uranium. This high-level waste would be separated into high- and low-activity fractions, which would be converted to glass (vitrified) in DWPF and to a cementitious low-level solid in the Saltstone Manufacturing and Disposal Facility, respectively. Recovered uranium could be sold to a commercial producer of nuclear fuel. DOE would dispose of the vitrified waste in a geologic repository and the saltstone in onsite vaults.

For Conventional Processing, DOE would use several existing SRS facilities:

- The L-Reactor Disassembly Basin and the Receiving Basin for Offsite Fuel for interim storage of the SNF before processing
- The F and H Canyons and related facilities for processing
- The high-level waste tank farms, DWPF, and Saltstone Manufacturing and Disposal Facility for high-level waste disposition

DOE expects that the Experimental Breeder Reactor-II fuel and the Mark-42 targets would be processed in F Canyon. The operation would result in the separation of plutonium that would be converted to metal in FB-Line and then placed in storage at SRS pending disposition in accordance with decisions reached under the *Surplus Plutonium Storage and Disposition EIS* currently being prepared by DOE. This material would not be used in any military application. All other processing operations would be conducted in H Canyon. Processing operations in H Canyon would continue if all fuel were to be processed until the aluminum-based SNF inventory was eliminated and the SNF receipt rate was low in about 2009 (i.e., receipts would be about 150 Materials Test Reactor-like elements per year and 12 High Flux Isotope Reactor assemblies per year). In parallel with processing operations, DOE could construct a Transfer, Storage, and Treatment Facility to receive and treat new SNF after processing operations cease. Because of the small volume of SNF to be processed in this facility, its dry storage capacity would be much less than required for other technologies.

Conventional Processing would be applicable to all fuel groups except most of the higher actinide targets (specifically the Mark-51 and "other" targets) and the non-aluminum-clad fuels. Conventional Processing would apply to the Mark-18s in the Higher Actinide Targets fuel group. The Record of Decision for the Programmatic SNF EIS (DOE 1995b) designated the Idaho National Engineering and Environ-

mental Laboratory as the location for management of non-aluminum-clad SNF. The SRS would store these fuels pending shipment to the Idaho National Engineering and Environmental Laboratory.

The resulting low-enriched uranium would not be suitable for use in weapons and any plutonium separated from the Experimental Breeder Reactor-II fuel or Mark-42 targets would be part of the plutonium considered surplus to the nuclear weapons program that will be dispositioned through decisions reached under the plutonium disposition EIS. Repository acceptance criteria should not be an issue because the vitrified high-level waste would be the same as the vitrified waste DOE is currently producing at SRS, and DOE has a high level of confidence that vitrified waste will meet the repository acceptance criteria. This option would add to the inventory of waste stored at SRS. However, sufficient storage and DWPF capacity exist to accommodate the added volume.

Figure 2-5 shows the Conventional Processing option. Appendix A provides more information on the technology.

2.3 Spent Nuclear Fuel Management Facilities

The implementation of the proposed action would require the construction of a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility and the use of several existing facilities, depending on the alternative selected. Table 2-4 lists the facilities required for the technologies. The following sections describe the existing and new facilities.

2.3.1 EXISTING FACILITIES

The existing SRS facilities that DOE would need for the proposed action are the L-Reactor Facility, the Receiving Basin for Offsite Fuel, and the F and H Canyons. Figure 2-6 shows the locations of these facilities. Appendix B provides information on the status of identified vulnerabilities at these facilities.

2.3.1.1 L-Reactor Facility

Facility Description

The Federal Government built L Reactor in the early 1950s to produce nuclear materials for national defense. In 1988 DOE shut the reactor down for safety upgrades, and has not restarted it. In 1993 the Department ended the reactor's materials production mission. The current mission of this facility is to store reactor components and other radioactive materials in the disassembly basin, receive and store foreign and domestic research reactor fuel in the disassembly basin, decontaminate shipping casks in the stack area, store contaminated moderator in tanks or drums, and compact low-level waste in a compactor. DOE maintains the structures, systems, and components necessary to perform these missions, but has deenergized, drained, or otherwise deactivated many others.

In addition to the support systems, L Reactor has three principal areas that could be important to the proposed action – the disassembly basin, the L-Reactor building, and the stack area. Figure 2-7 shows L-Reactor and indicates the locations of these areas.

The disassembly basin, which would be the principal structure supporting the SNF storage mission, is a large concrete basin containing approximately 3.4 million gallons (13,000 cubic meters) of water varying in depth from 17 to 50 feet (5.2 to 15 meters). DOE has upgraded the basin to improve water control and monitoring, including continuously operating deionizers to improve water chemistry, makeup water deionizers, and a water level monitoring system. In addition, DOE has added storage racks to accommodate anticipated fuel receipts. The disassembly basin contains a transfer bay with one water-filled pit and heavy lifting equipment to transfer shipping casks to the basin.

The L-Reactor building has space potentially suitable for installation of facilities for treatment of SNF (see Section 2.3.2.2). The space includes the process room and crane maintenance area. The process room, a shielded area situated

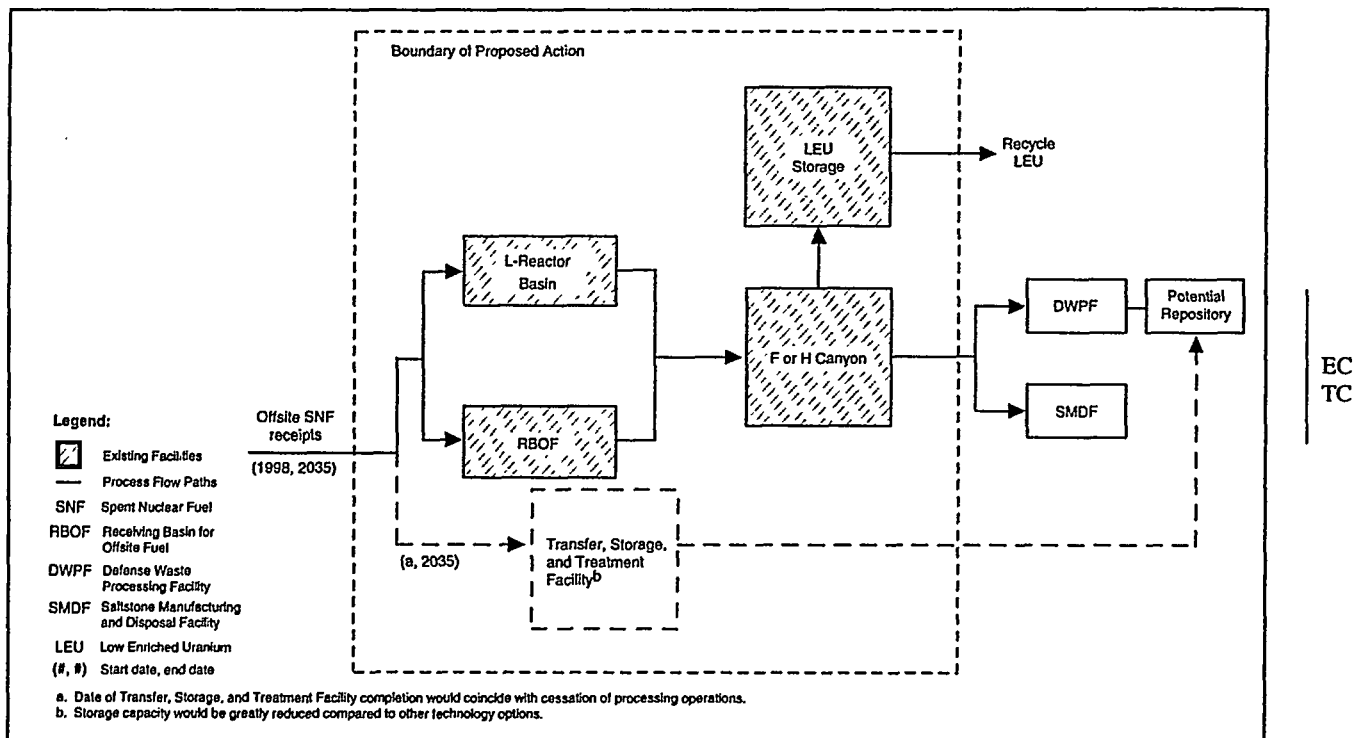


Figure 2-5. Conventional Processing.

Table 2-4. Facilities needed for SNF technologies.

Technology	Receiving Basin for Offsite Fuel	L-Reactor Facility	F or H Canyon	Transfer and Storage Facility	Melt and Dilute Treatment Facility	Mechanical Dilution Treatment Facility	Vitrification Facility	Electrometallurgical Treatment Facility	Renovated Reactor Facility
1. Prepare for Direct Disposal/Direct Co-Disposal	✓	✓		✓					✓
2. Repackage and Prepare to Ship ^a	✓	✓ ^b		✓					✓
3. Melt and Dilute	✓	✓		✓	✓				✓
4. Mechanical Dilution	✓	✓		✓		✓			✓
5. Vitrification Technologies	✓	✓		✓			✓		✓
6. Electrometallurgical Treatment	✓	✓		✓				✓	✓
7. Conventional Processing	✓	✓	✓	✓	✓ ^c				
8. Continued Wet Storage	✓	✓							

a. To another DOE site.
b. Needed only if a Transfer, Storage, and Treatment Facility were implemented in a reactor facility.
c. Once conventional processing is terminated, the remaining SNF would require treatment using one of the new technologies. A Melt and Dilute Treatment Facility is included as part of Conventional Processing as a reference follow-on treatment

above the reactor tank, formerly provided access to the reactor by means of a charge and discharge machine for handling reactor fuel assemblies. The area is serviced by an overhead crane. Fuel assemblies were transferred from the L-Reactor Disassembly Basin to the process room by way of an interconnecting water canal. The crane maintenance area, connected to the process room by a shielded crane wash area, allowed hands-on maintenance of the fuel assembly transfer systems.

DOE uses the L-Reactor stack area to unload shipping casks from their International Organization for Standardization (ISO) containers and to decontaminate empty shipping casks. The decontamination hut has a sump pump, spray equipment, a ventilation system, and deionizers.

In 1993 DOE performed a vulnerability assessment of its SNF facilities and identified several vulnerabilities related to the disassembly basins (DOE 1993). The Defense Nuclear Facilities Safety Board reported other vulnerabilities (DNFSB 1994; Burnfield 1995; Conway 1996), including the lack of adequate water chemistry control, which resulted in the corrosion of stored SNF and some cladding failure. The corroding fuel resulted in a buildup of radionuclides in the water and in the sludge at the bottom of the basins. Another vulnerability was the lack of an adequate leak detection capability. Since the vulnerability assessments, DOE has completed the corrective actions. One of the more significant upgrades is the installation of deionizers for maintaining water quality; maintenance of water chemistry is important to minimize corrosion. Appendix B describes these vulnerabilities and corrective action plans in greater detail.

Facility Operations

DOE would receive SNF in shipping casks designed to meet SNF cask design criteria (10 CFR 71). If the cask was too large for the L-Reactor Disassembly Basin or if other operational restrictions (such as a maintenance outage) occurred, DOE would transport the cask to the Receiving Basin for Offsite Fuel in H Area,

remove the fuel and place it in a smaller cask, and transfer it to L Reactor. The smaller casks would be moved to the transfer bay of the disassembly basin.

SNF is unloaded from the casks underwater. The procedure is as follows: the casks are vented, filled with water, and submerged in the transfer bay. The purged air is cleaned by high-efficiency particulate air filters before being discharged to the atmosphere. The casks are opened and the fuel elements placed in a bucket for examination. If the fuel cannot be identified or is inconsistent with the documentation provided by the reactor operator, it is isolated until the discrepancy is resolved.

The SNF is moved to the storage area of the disassembly basin through a transfer canal. The cask lid is replaced and the cask is drained, washed, and decontaminated. Decontamination water is sent to the disassembly basin.

2.3.1.2 Receiving Basin for Offsite Fuel

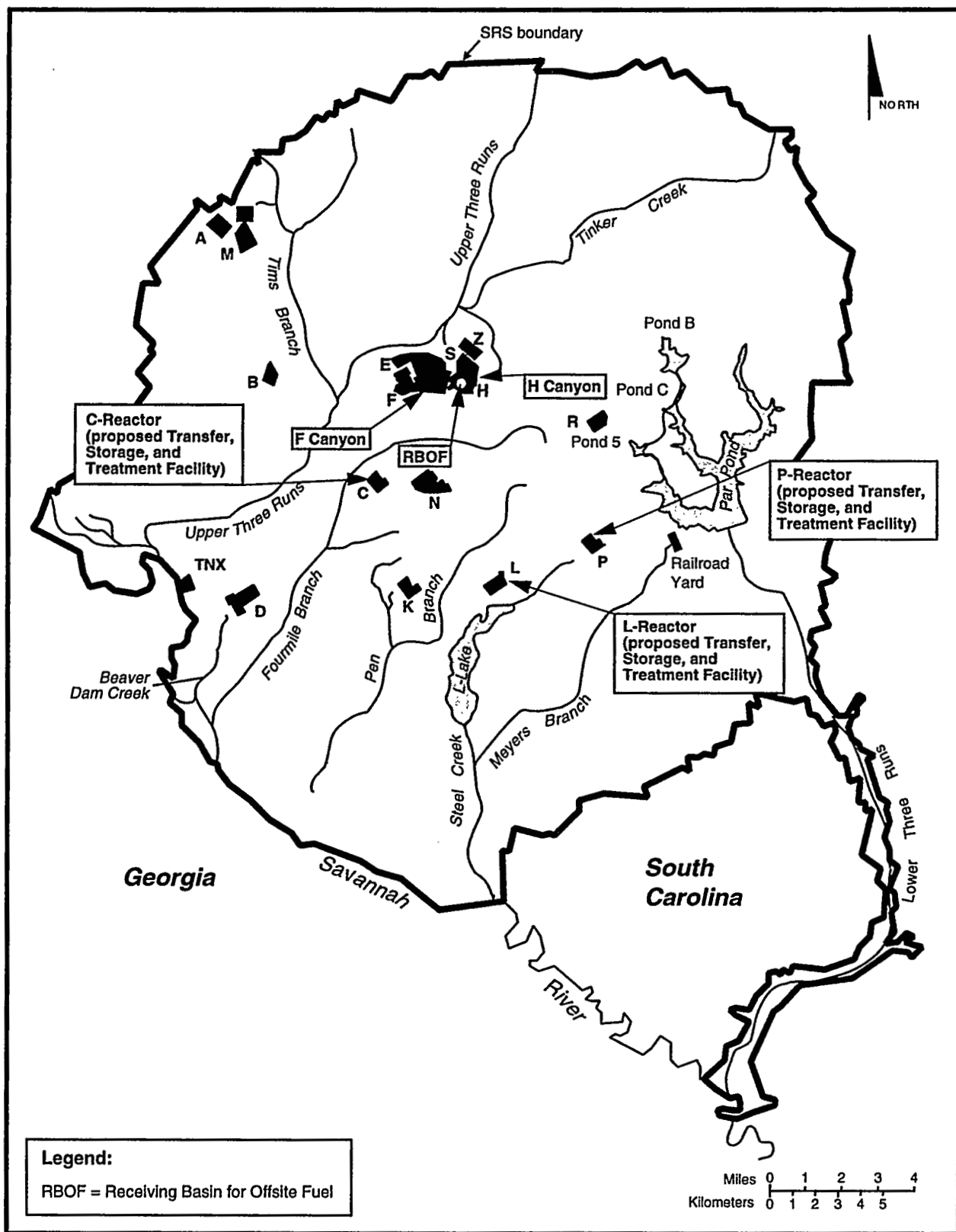
Facility Description

The Receiving Basin for Offsite Fuel, located in H Area, has provided storage for irradiated SNF since 1964. It has an unloading basin, two storage basins, a repackaging basin, a disassembly basin, and an inspection basin, all underwater. Fuel is handled or stored under at least 4 feet (1.2 meters) of water to provide shielding against radiation. The reinforced-concrete basins are below grade. They have either chemical coatings or stainless-steel linings for ease of decontamination. The storage lattice in the basins consist of rows of racks of aluminum I-beams. Gratings, guide plates, and spacers between the racks separate individual storage positions and provide the spacing required for criticality safety.

In addition to the water-filled basins, the Receiving Basin for Offsite Fuel has a receiving bay, dry cask inspection pit, control room, office areas, equipment storage areas, and concrete cells that contain tanks for water decontamination (deionization) and temporary storage of

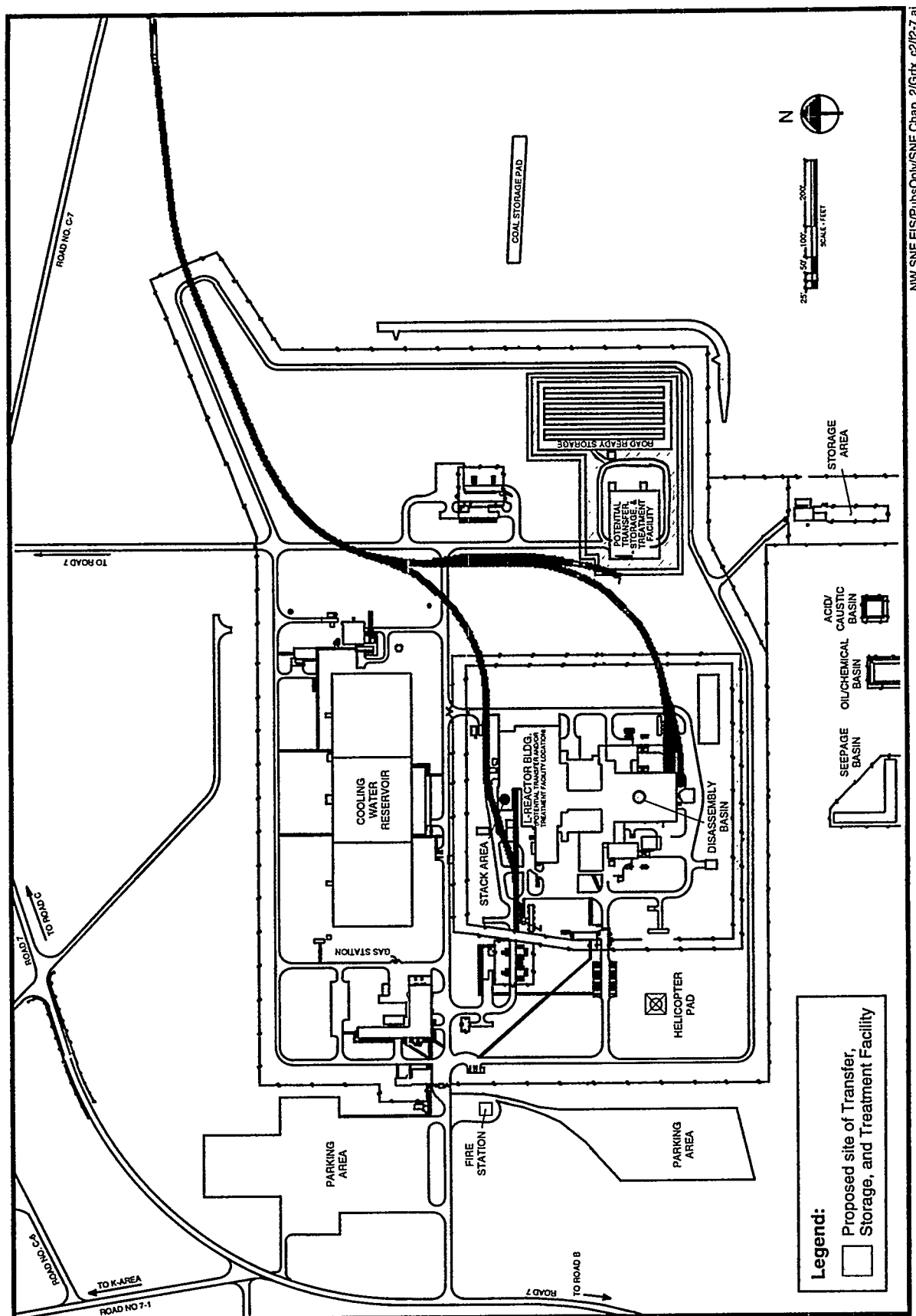
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Figure 2-6. SRS map indicating locations of facilities needed for Proposed Action.



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Figure 2-7. Plan view of the L-Reactor facility.

radioactive liquid waste. The facility has a 100-ton (91-metric-ton) bridge crane that travels on rails approximately 31 feet (9 meters) above grade. The crane has two 50-ton (45-metric-ton) hoists and two 3-ton (2.7-metric-ton) hoists. The crane travels over the cask receiving, unloading, and fuel storage areas.

The DOE vulnerability assessment (DOE 1993) and inspections performed by the Defense Nuclear Facilities Safety Board (Burnfield 1995; Conway 1996) identified vulnerabilities related to the Receiving Basin for Offsite Fuel. These vulnerabilities primarily involved the seismic qualification of the building, the lack of adequate leak detection, and the spacing of vertically stored fuel assemblies (a criticality concern). Appendix B describes these vulnerabilities and their corrective actions (which have all been completed).

Facility Operations

The receiving bay on the north side of the Receiving Basin for Offsite Fuel receives shipping casks containing irradiated fuel delivered by truck or rail. Radiological surveys of the casks determine external radiation and surface contamination levels. The cask is vented after cleaning and filled with water that is sampled to detect contamination, which would indicate damaged or failed fuel. The cask lid bolts are loosened and the cask transferred to the cask basin using the 100-ton (91-metric-ton) overhead crane. The cask is lowered into the basin until the top of the lid is approximately 3 feet (1 meter) above the water surface and the lid bolts are removed. The cask is lowered to the bottom of the basin and the lid removed. Fuel elements are removed from the cask and placed in transfer buckets, cans, or bundles, depending on the fuel design. The bucket, can, or bundle is placed in a storage rack and the process repeated until all fuel had been unloaded from the cask.

The Receiving Basin for Offsite Fuel has separate basins to segregate and can damaged or failed fuel, disassemble fuel components by mechanical means (e.g., cutting), or perform inspection and measurement. The basin water

circulates through a filter and a deionizer for purification and clarification. DOE replaces the filters and deionizers periodically, depending on radioactivity or impurity levels in the water. "

2.3.1.3 F and H Canyons

Facility Description

Two SRS facilities – F and H Canyons – could chemically separate uranium from fission products in SNF. The canyon facilities are nearly identical and use similar radiochemical processes for the separation and recovery of plutonium, neptunium, and uranium isotopes. Historically, F Canyon recovered plutonium-239 and uranium-238 from irradiated natural or depleted uranium, and H Canyon recovered plutonium-238, neptunium-237, and uranium-235 from irradiated reactor fuels and targets.

The canyons buildings are reinforced-concrete structures, 835 feet (254 meters) long by 122 feet (37 meters) wide by 66 feet (20 meters) high. They house the large equipment (tanks, process vessels, evaporators, etc.) used in the chemical separations processes.

Each canyon facility contains two canyons, the hot canyon and the warm canyon. The two canyons are parallel and separated by a center section, which has four floors. The center section contains office space, the control room for facility operations, chemical feed systems, and support equipment such as ventilation fans. Processing operations involving high radiation levels (dissolution, fission product separation, and high-level radioactive waste evaporation) occur in the hot canyon, which has thick concrete walls to shield people outside and in the center section from radiation. The final steps of the chemical separations process, which generally involve lower radiation levels, occur in the warm canyon. The F and H Canyons are designed to prevent the release of airborne radioactivity. The ventilation systems maintain a negative air pressure with respect to outside pressure. The ventilation discharges are filtered by high-efficiency particulate air filters and sand filters that remove more than 99.9 percent of the

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particulate radioactivity. Figure 2-8 shows a cutaway view of a canyon building. Figure 2-9 is an aerial photograph of H Canyon and the surrounding area.

DOE and the Defense Nuclear Facilities Safety Board have identified environmental, safety, and health vulnerabilities at the F and H Canyons (DOE 1993; DNFSB 1994). These vulnerabilities relate to the seismic qualification of the buildings and the continued storage of in-process nuclear materials. DOE has verified the seismic qualification of the canyons. In accordance with the various Records of Decision for the Interim Management of Nuclear Materials EIS (DOE 1995a), DOE is stabilizing selected materials of concern identified by the Defense Nuclear Facilities Safety Board.

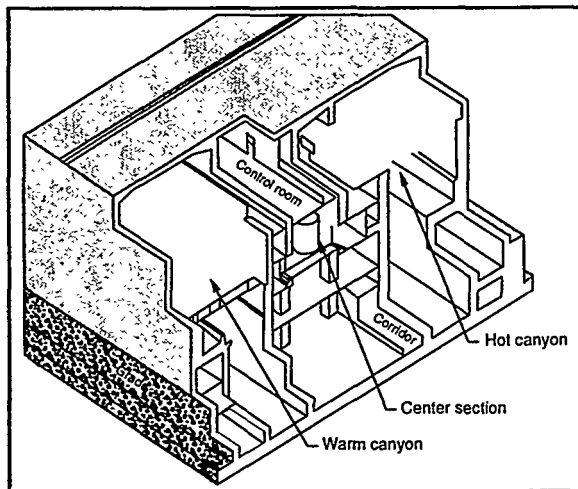


Figure 2-8. Canyon building sections.

Facility Operations

The SNF would arrive by rail in a shielded shipping cask from either the Receiving Basin for Offsite Fuel or the L-Reactor Disassembly Basin. The fuel would be unloaded and placed in an interim storage pool by a remotely operated crane. At the appropriate time, the fuel would be placed in the dissolver and dissolved by nitric acid. If the processing was performed in F Canyon, the acid solution would be blended down with depleted uranium. However, because H Canyon is designed to handle enriched uranium, the blending to low enriched uranium in

H Canyon could occur at virtually any point in the processing operation. In either case, the uranium would be blended to about 5 percent uranium-235.

The resulting acid solution would be chemically processed using clarification and solvent extraction to produce a relatively pure and concentrated stream of uranyl nitrate, which would be stored in tanks awaiting disposition including selling it to commercial reactor fuel users/manufacturers. Building ventilation discharge would be filtered (including sand filters) to remove at least 99.9 percent of the particulate radioactivity.

2.3.2 Proposed Facilities

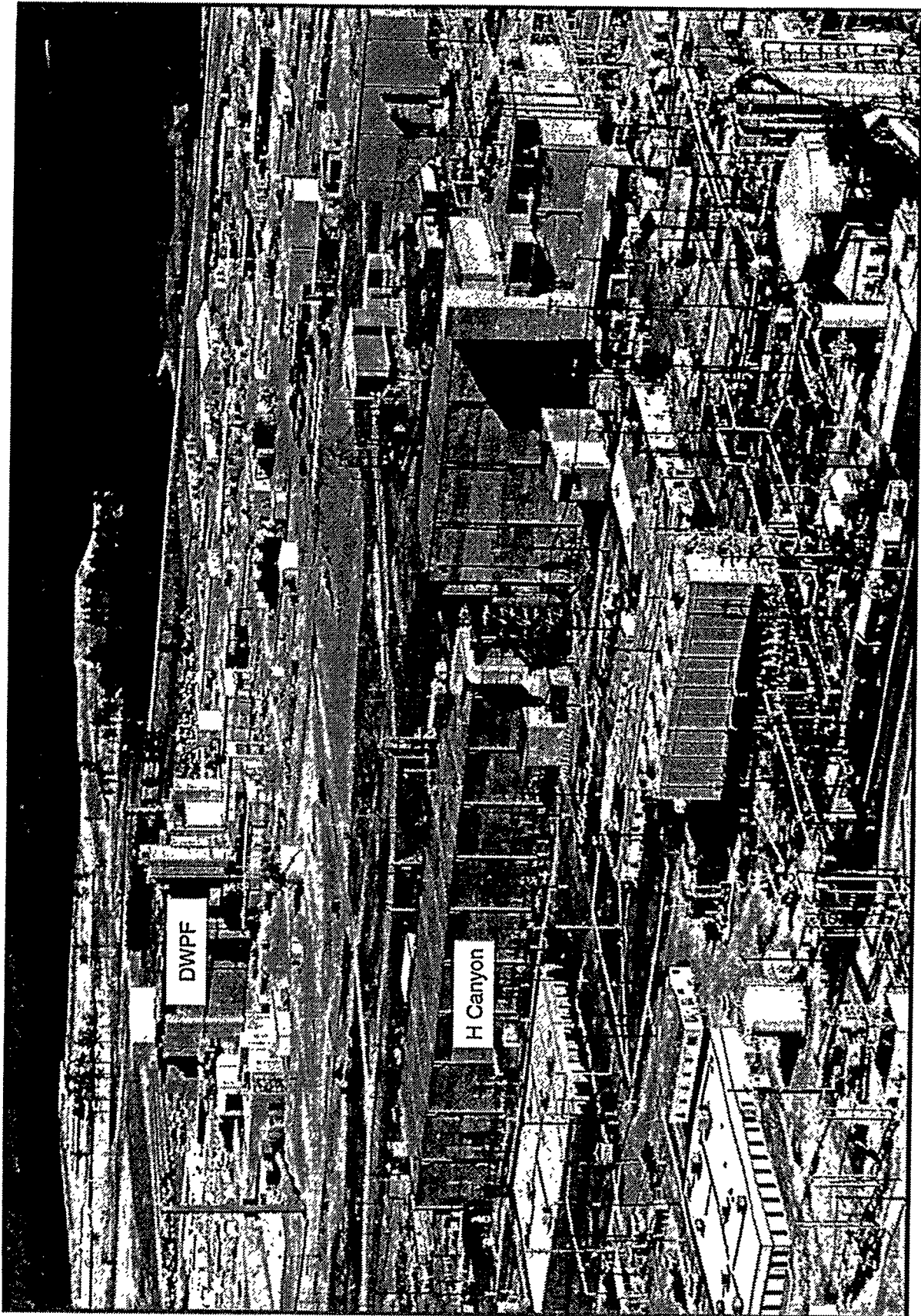
DOE could construct new facilities or modify existing ones to accomplish the Proposed Action, depending on the alternative selected.

2.3.2.1 Transfer and Storage Facility

A Transfer and Storage Facility would provide remote handling and heavy lifting capability, hot cells, and space to receive SNF shipments. This facility would place SNF in interim storage as needed, open the shipping containers, sample and analyze the fuel, crop end fittings if necessary, vacuum-dry the SNF, repackage the fuel in storage canisters, and place the repackaged fuel in interim storage. DOE would use this facility to perform the functions listed in Table 2-5 without the use of water-filled storage pools; however, DOE could choose to provide the capability to receive incoming SNF in a wet basin. This small wet basin, if used, would be for receipt only - not storage. Figure 2-10 shows this facility.

The dry storage segment of the facility would provide lag storage for SNF waiting for preconditioning or treatment, road-ready storage for fuel packaged for shipment to a geologic repository, and temporary storage for empty canisters and loaded and unloaded transportation casks. The size of the storage facility would depend on how DOE decided to implement the Proposed Action. For example, if DOE

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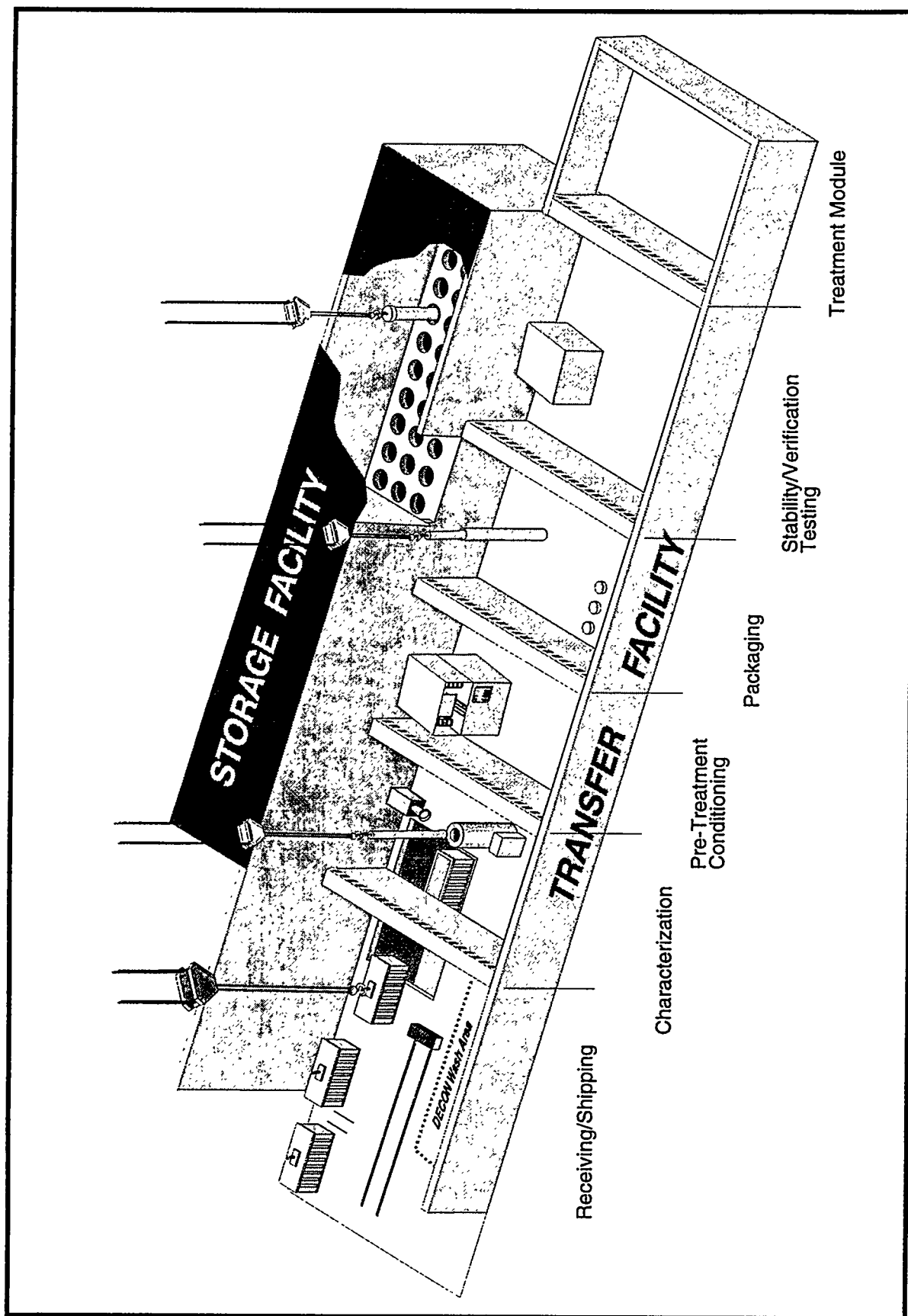


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Figure 2-9. H Canyon and surrounding area (view toward northeast).

Table 2-5. Transfer and Storage Facility functions.

Function	Description
Receiving/shipping	Receive casks, unload SNF, load casks, and prepare loaded and unloaded casks for shipment
Characterization	Inspect SNF for storage, conditioning, and disposition (e.g., visual inspection, gamma spectrometry, and calorimetry)
Conditioning	Crop end fittings or binding pins; activity would not breach cladding or modify the fuel matrix
Packaging	Place SNF in appropriate cans and canisters (e.g., vacuum drying, filling with inert gas) and packaging for road-ready storage or direct transport
Stability/verification testing	Provide analytical capabilities to perform sampling and analysis to verify conformance to repository waste acceptance criteria
Treatment Facility Interface	Provide interfaces necessary to accommodate various treatment technologies
Storage	Provide dry road-ready storage using modular design and construction



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Figure 2-10. Schematic cut-a-way of the Transfer, Storage, and Treatment Facility.

selected Electrometallurgical Treatment as a new processing technology, the storage component of the facility would only need to provide lag storage for fuel awaiting treatment; no road-ready storage would be necessary because waste produced from the Electrometallurgical Treatment would be sent to DWPF. Table 2-6 lists the number of road-ready canisters DOE would need to store for each technology. In each case, the number of canisters for the treatment technologies is less than that for the Direct Co-Disposal technology. The size of the transfer operations component of the facility would be independent of any new technology selected. In the event Conventional Processing is implemented, the size of the Transfer and Storage Facility would be reduced by about 30 to 60 percent.

The storage segment probably would have one of the three generic designs shown in Figure 2-11. Regarding the environmental impacts of constructing and operating a dry storage fa-

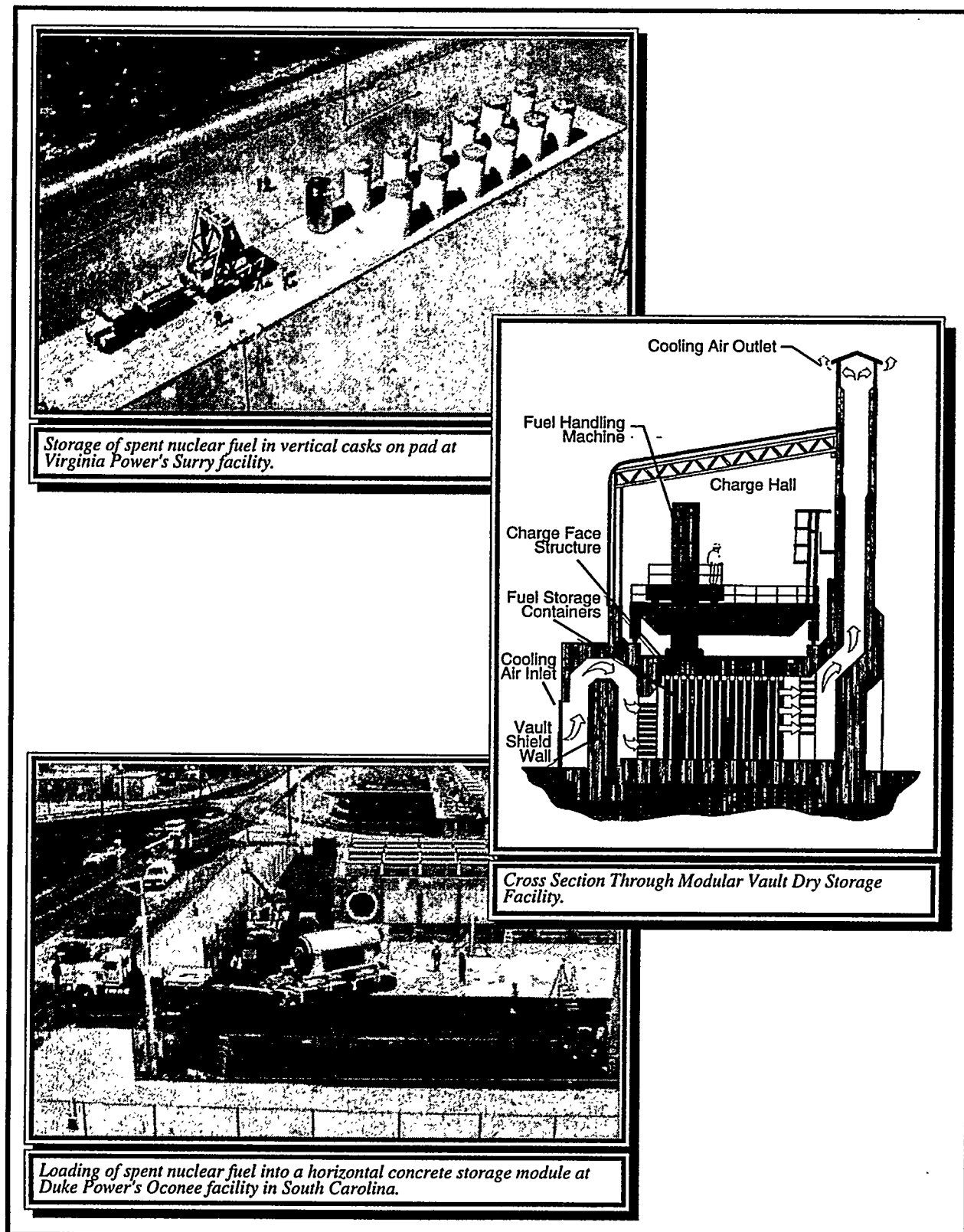
cility, the *Foreign Research Reactor Spent Nuclear Fuel EIS* (DOE 1996c) concluded, "There are significant differences between these technologies in terms of construction, operations and maintenance costs and various design details. However, these differences do not result in any important variations in environmental impacts and consequences."

The modular dry storage vault design is a self-contained concrete structure that would provide storage for hundreds of SNF assemblies. The vault would contain a charge and discharge bay with an SNF-handling machine above a floor containing steel tubes to house the removable fuel canisters. The bay would be shielded from the stored fuel by the thick concrete floor and shield plugs inserted at the top of the steel storage tubes. Large labyrinth air supply ducts and discharge chimneys would permit natural convection cooling of the fuel storage tubes to dissipate decay heat. The perimeter concrete walls would provide shielding.

Table 2-6. Road-ready storage capacities.

Technology	Number of co-disposal canisters (17-inch diameter)
Prepare for Direct Co-Disposal/Direct Disposal	1,400 ^a
Repackage and Prepare to Ship	0
Melt and Dilute	400
Mechanical Dilution	630
Vitrification Technologies	1,350 ^b
Electrometallurgical Treatment	0 ^c
Conventional Processing	0 ^d
Continued Wet Storage	0

a. Direct Disposal in 24-inch diameter canisters would require 1,100 canisters.
b. Vitrification Technologies would produce 24-inch diameter canisters. The value reported is for Dissolve and Vitrify and Glass Material Oxidation and Dissolution System. Plasma Arc Treatment would produce 490 24-inch diameter canisters.
c. Electrometallurgical treatment would produce about 90 high-level waste canisters to be stored in the Glass Waste Storage Building of the Defense Waste Processing Facility.
d. Conventional Processing would result in storage of about 150 high-level waste canisters in the Glass Waste Storage Building of the Defense Waste Processing Facility.



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Figure 2-11. Typical spent nuclear fuel dry storage facilities.

A dry concrete storage cask, either vertical cask-on-pad or a horizontal concrete module, would perform a similar function, but would not be in a vault. The cask would provide the shielding. A dedicated truck and trailer would transport the fuel containers from the transfer area of the facility to the dry storage area. A ram (for horizontal modules) or a crane (for vertical modules) would insert the fuel package into the storage cask. Appendix F of the *Foreign Research Reactor Spent Nuclear Fuel EIS* (DOE 1996c) contains more information on dry storage facility designs.

DOE used a formal site selection process (Wike et al. 1996) to identify and evaluate potential sites for the construction of the Transfer and Storage Facility. Among the siting criteria were engineering and operational parameters; infrastructure support; human health, environmental, and ecological impacts; regulatory criteria; and land use planning. The process identified five potential sites, two of which received substantially higher scores than the others. These sites are the east side of L Area inside the facility fence, and the southeast side of C Area inside the facility fence. DOE has determined that these two sites are preferred and has completed some geotechnical evaluations on them. Figures 2-7 and 2-12, respectively, show these locations. DOE has considered these two sites in the analyses in this EIS. The transfer functions performed by a Transfer and Storage Facility could also be located in a renovated reactor building. Storage facilities would be as described above.

2.3.2.2 Transfer, Storage, and Treatment Facility

DOE could build a new Transfer, Storage, and Treatment Facility in the locations previously described for the Transfer and Storage Facility. Alternatively, the facility could be located in a new facility in F or H Area (Figures 2-13 and 2-14) to take advantage of existing services and infrastructure in these areas. DOE would con-

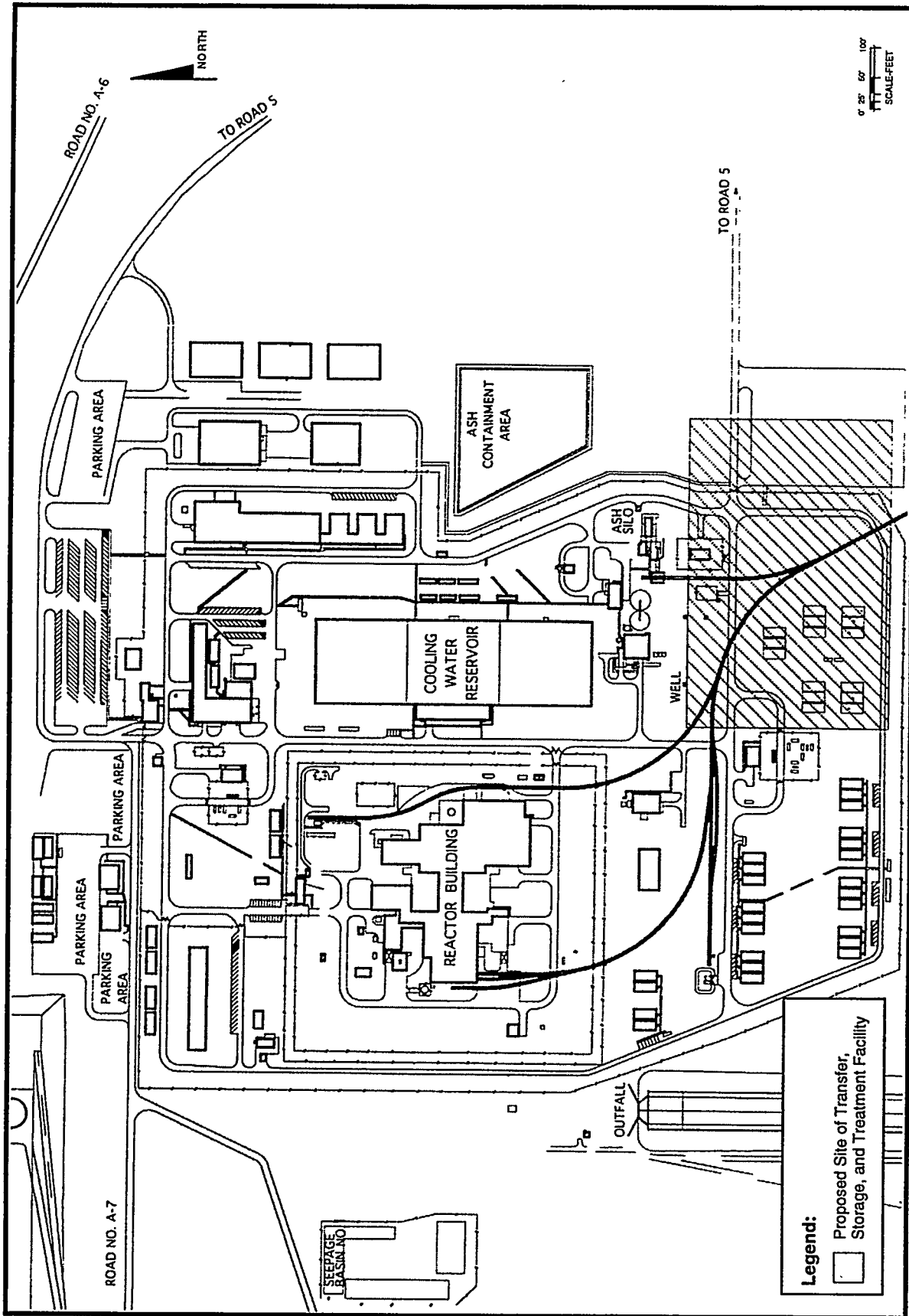
struct this facility only if it selected a technology that required it. The facility would be similar to the Transfer and Storage Facility described in Section 2.3.2.1, but with the addition of SNF treatment capability as described in the following paragraphs. The operations performed in the facility would depend on the treatment technology DOE selected, and could include Melt and Dilute, Mechanical Dilution, Vitrification Technologies, or Electrometallurgical Treatment.

The facility design would address criticality issues during normal operations and under conditions of extreme natural phenomena. The facility would contain hot cells, remote handling equipment for the fuel and canisters, processing equipment such as melters (depending on the technology option selected), waste handling and treatment capability, canister decontamination capability, and infrastructure needed for radiological protection operations (e.g., monitoring equipment and protective clothing change rooms). Treatment and handling operations would be performed in facility areas especially designed to prevent the release of airborne radioactivity. For example, the ventilation system would maintain a negative air pressure with respect to outside pressure. The ventilation discharge would be filtered to remove at least 99.9 percent of the particulate radioactivity.

DOE also is considering performing SNF treatments in a renovated reactor facility. In this EIS, DOE has evaluated modifying Building 105-L, and DOE considers this evaluation representative of other reactor area facilities. The processes for transfer and treatment would be located within the L-Reactor building (Figure 2-7), supported by capabilities in the existing structure and adjacent L-Area enclosure. The treatment facilities would be operated in close conjunction with the underwater storage of the SNF in the L-Reactor Disassembly Basin, converting the SNF to the final waste form for dry storage in a Storage Facility as described in Section 2.3.2.1.

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Figure 2-12. Plan view of the C-Reactor facility.

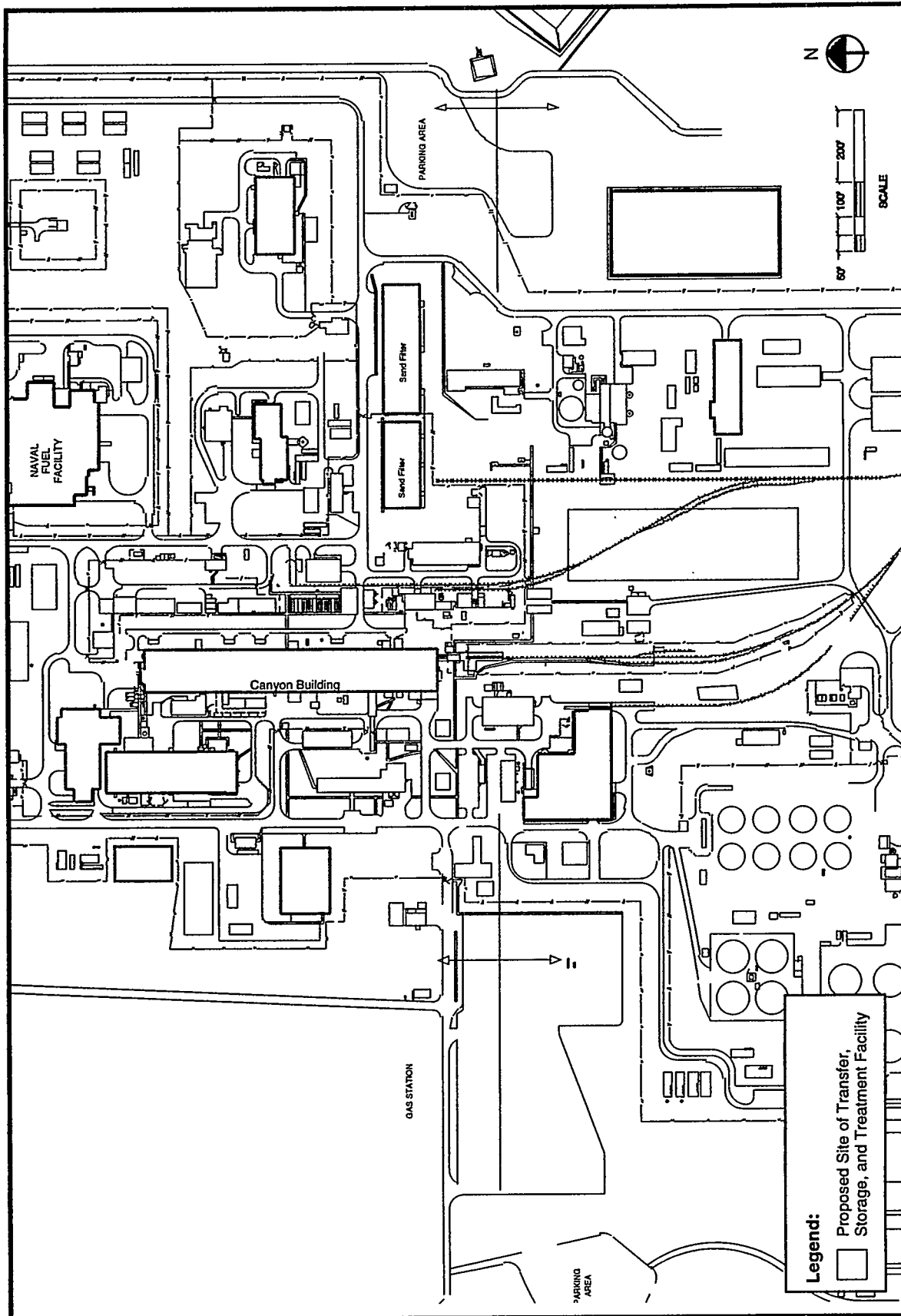
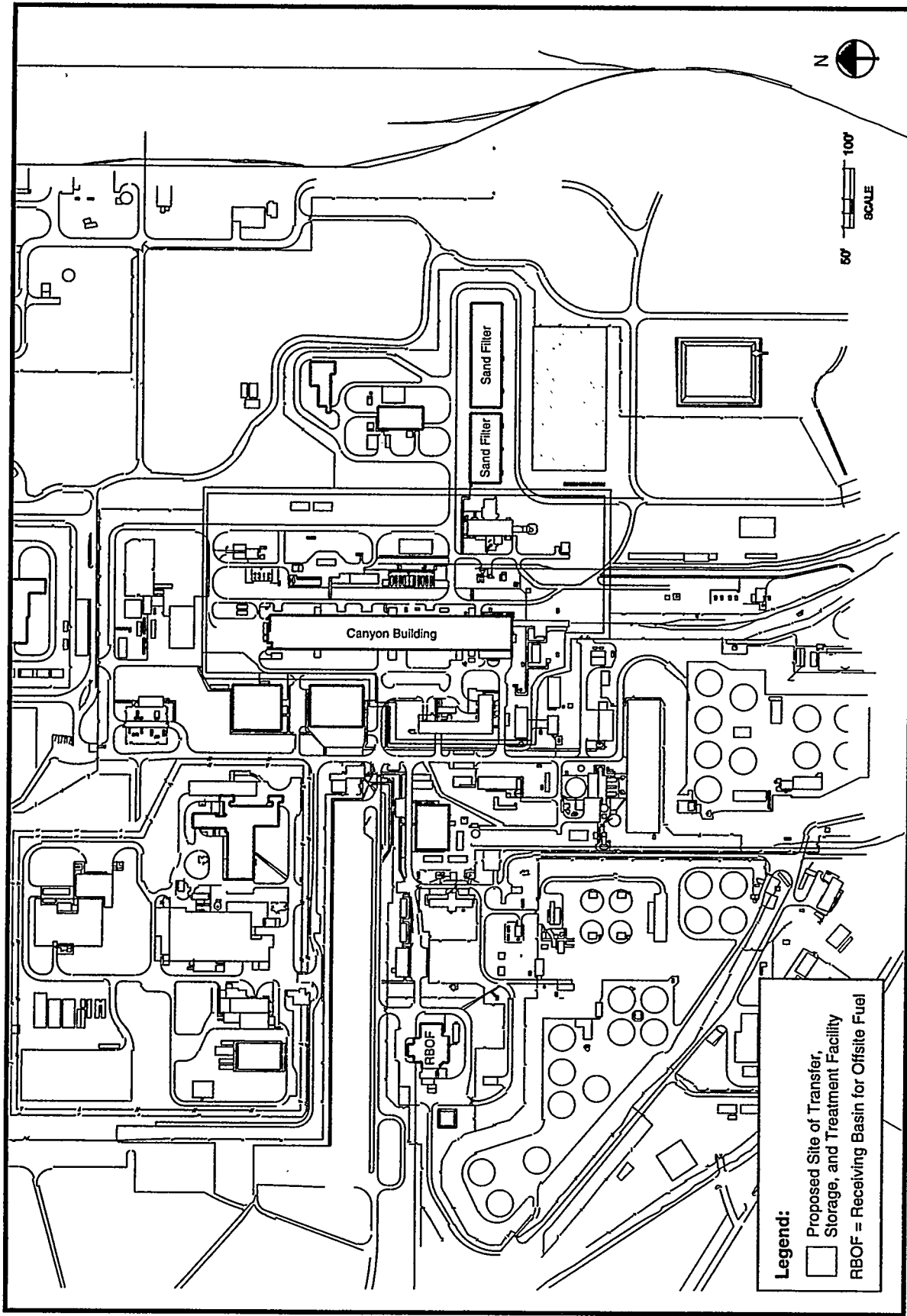


Figure 2-13. Potential Transfer, Storage, and Treatment Facility location in F Area.



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Figure 2-14. Potential Transfer, Storage, and Treatment Facility location in H Area.

Table 2-7. Fuel groups and technology options that could be applied to meet the purpose and need. For each fuel group, the technologies that would produce the lowest and highest impacts have been identified.

Fuel group	1.	2.	3.	4.	5.	6.	7.
	Prepare for Direct Co-Disposal	Repackage and Prepare to Ship ^a	Melt and Dilute	Mechanical Dilution	Vitrification Technologies	Electro-metallurgical Treatment	Conventional Processing
A. Uranium and Thorium Metal Fuels	Yes ^b , LB ^c	No	Yes	No	Yes	Yes	Yes, UB ^d
B. Materials Test Reactor-Like Fuels	Yes, LB	No	Yes	Yes	Yes	Yes	Yes, UB
C. HEU/LEU ^e Oxides and Silicides Requiring Resizing or Special Packaging	Yes, LB	No	Yes	Yes	Yes	Yes	Yes, UB
D. Loose Uranium Oxide in Cans	No	No	Yes, LB	No	Yes	Yes	Yes, UB
E. Higher Actinide Targets ^f	NA	Yes, LB/UB	NA	NA	NA	NA	NA
F. Non-Aluminum Clad Fuels ^f	NA	Yes, LB/UB	NA	NA	NA	NA	NA

a. This alternative describes repackaging for storage at SRS pending shipment offsite.

b. "Yes" indicates that the technology can be applied to the fuel group. "No" indicates that the technology cannot be applied to the fuel group.

c. LB = lower bound of impacts.

d. UB = upper bound of impacts.

e. HEU = highly enriched uranium; LEU = low enriched uranium.

f. NA = not applicable; not decided in this EIS. Higher actinide targets would be stored until DOE determined their disposition and non-aluminum clad fuel is scheduled to be shipped to Idaho National Engineering and Environmental Laboratory for treatment. Only the impacts of storing these materials are considered in this EIS.

2.4 Alternatives Evaluated

As indicated in Sections 2.2.1 through 2.2.3, none of the technologies is likely to be applicable to all the fuel groups. Table 2-7 lists the technology options DOE believes are applicable to the fuel groups discussed in this EIS. DOE probably would implement a combination of options to accomplish SNF management at SRS. Many (more than 700) technology-fuel group configurations can be created using the information in Table 2-7. Tables 2-1 and 2-2 summarize the basis for the applicability of the New Packaging options and the New Processing Technology options. Conventional Processing could be applied to any fuel group except the non-aluminum-clad fuels and the higher actinide targets. Although the No-Action Alternative could be applied to all fuel groups, it would not meet the purpose and need for action.

Taking into consideration the technology options available to the various fuel groups and decisions previously made about managing certain types of SNF, DOE developed five alternatives to analyze in this EIS. DOE has chosen to present impacts from the No Action Alternative, the Preferred Alternative, the Direct Disposal Alternative, and the Maximum- and Minimum-Impact Alternatives described below to illustrate the range of impacts that could occur from any configuration the decisionmakers might select (Table 2-8). These configurations are representative of the range of those DOE could select to accomplish the proposed action and are expected to include the upper and lower bounds of potential impacts. The No Action Alternative represents the impact from current operations.

DOE recognizes that a combination of technology options might not result in the lowest or highest impact for all evaluated technical parameters (e.g., for a particular configuration, worker health and public health impacts could be lowest, but radioactive waste generation could be highest) and that there are other reasonable alternative configurations that would result in similar minimal or substantial impacts. Impacts resulting in human health effects and

environmental pollution received greater weight than those resulting in the consumption of natural resources or waste disposal space. In addition, impacts to the general public received greater weight than those to SRS workers. Similarly, impacts that would occur immediately (e.g., operation of new and existing processing facilities) received greater weight than impacts that are not expected but could occur in the distant future.

2.4.1 MINIMUM IMPACT ALTERNATIVE

This alternative consists of the fuel groups and technologies that DOE believes would result in the lowest overall impact. The identification of the minimum impact (and environmentally preferred) alternative required both quantitative and qualitative analyses. The first step tabulated the analytical parameters (e.g., volume of high-level waste, air concentrations) and the minimum-impact technology for each parameter for each fuel group. The selected analysis parameters often resulted in a combination of high and low impacts for a particular fuel group. Therefore, the second step required a qualitative examination of trends in combinations that would provide overall minimum impacts.

DOE believes that the range of impacts from other reasonable choices of the minimum-impact alternative would be small. Therefore, DOE expects that the impacts of this alternative would be representative of the lower bound of impacts from the Proposed Action.

The minimum impact alternative would include New Packaging and New Processing Technologies options. Material Test Reactor-like fuels and highly enriched uranium/low enriched uranium (HEU/LEU) oxides and silicides would be treated using the Direct Disposal/Direct Co-Disposal option and placed in the Transfer and Storage Facility with a minimum of treatment (e.g., cold-vacuum drying and canning). The uranium and thorium metal fuels would be treated using the Direct Disposal/Direct Co-Disposal option but more rigorous treatment (i.e., hot-vacuum drying) would be required.

Table 2-8. Alternatives analyzed in this EIS.

Fuel Group	No-Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative
A. Uranium and Thorium Metal Fuels	Continued Wet Storage	Prepare for Direct Co-Disposal	Conventional Processing	Conventional Processing	Conventional Processing
B. Materials Test Reactor-like Fuels	Continued Wet Storage	Prepare for Direct Co-Disposal	Prepare for Direct Co-Disposal	Melt and Dilute	Conventional Processing
C. HEU/LEU Oxide and Silicides Requiring Resizing or Special Packaging	Continued Wet Storage	Prepare for Direct Co-Disposal	Prepare for Direct Co-Disposal	Melt and Dilute ^a	Conventional Processing
D. Loose Uranium Oxide in Cans	Continued Wet Storage	Melt and Dilute	Melt and Dilute ^b	Melt and Dilute ^b	Conventional Processing
E. Higher Actinide Targets	Continued Wet Storage	Repackage and Prepare to Ship to Another DOE Site	Repackage and Prepare to Ship to Another DOE Site ^c	Continued Wet Storage	Repackage and Prepare to Ship to Another DOE Site ^c
F. Non-Aluminum-Clad Fuels	Continued Wet Storage	Repackage and Prepare to Ship to Another DOE Site	Repackage and Prepare to Ship to Another DOE Site	Repackage and Prepare to Ship to Another DOE Site	Repackage and Prepare to Ship to Another DOE Site

- a. Conventional processing would be the preferred technology for the failed or sectioned Oak Ridge Reactor fuel, High Flux Isotope Reactor fuel, Tower Shielding Reactor fuel, Heavy Water Components Test Reactor fuel, and a Mark-14 target.
- b. Conventional processing is the preferred technology for the Sterling Forest Oxide fuel.
- c. Conventional processing is the applicable technology for the Mark-18 target assemblies (approximately 1 kilogram heavy metal), under these two alternatives.

(DOE notes there is a high degree of technical uncertainty regarding the acceptability of this material in a repository; however, Direct Co-Disposal was postulated to represent minimum impacts.)

TC DOE would continue to wet store the Mark-51 and other Higher Actinide Targets at the SRS. Additionally, DOE would continue to wet-store the non-aluminum-clad spent nuclear fuel at SRS until the material is shipped to the Idaho National Engineering and Environmental Laboratory. In the event the non-aluminum clad fuel have not been transferred offsite by the time a dry storage facility is in operation at the SRS (to support the Melt and Dilute Technology), DOE could repackage the fuel and transfer the material to dry storage. To maintain operational flexibility, DOE could transfer the Mark-51 and other targets to dry storage. DOE would maintain the Mark-18 targets in wet storage pending disposition decisions due to potential health and safety concerns associated with the actions that would be required to repackage the Mark-18 target assemblies.

TC While in wet storage, if fuel began to deteriorate, resulting in imminent environmental, safety, and health vulnerabilities, DOE would use the canyons, if they were operating, to stabilize the vulnerable materials.

TC The loose uranium oxide in cans would not be contained in a tightly bound matrix and, therefore, may not be acceptable for placement in a geologic repository. Therefore, the Melt and Dilute technology would be used to treat these fuels.

2.4.2 MAXIMUM IMPACT ALTERNATIVE

This alternative provides the upper bound on the range of impacts from potential configurations. It would provide conventional processing for all SNF except the higher actinide targets and the non-aluminum-clad fuels selected for offsite shipment.

DOE expects that the Experimental Breeder Reactor-II and Mark-42 targets from the uranium and thorium metal fuels group would be processed in F Canyon. All other processing operations would be conducted in H Canyon. Processing operations in H Canyon would continue until the aluminum-based SNF inventory was eliminated and the SNF receipt rate was low (i.e., about 150 Materials Test Reactor-like elements per year and 12 High Flux Isotope Reactor assemblies per year; approximately 2009). In parallel with processing operations, DOE could construct a Transfer, Storage, and Treatment Facility with treatment capability to receive and treat new SNF after processing operations cease. Once the Transfer, Storage, and Treatment Facility was completed, processing in the canyons would be phased out.

Analyses of the maximum impact alternative are conservative in that they assume that the entire SNF inventory would be processed in the canyons, which would produce the greatest impacts of all the treatment options. No credit is taken for discontinuing use of the canyons and processing some of the inventory in a new treatment facility.

Although this EIS proposes only to continue to store Mark-18 targets, DOE has included the impacts of processing the Mark-18 targets in the Maximum Impact Alternative. The analysis of impacts is taken from the Final Environmental Impact Statement for Interim Management of Nuclear Materials. The 12-foot long Mark-18 targets would require size reduction for transport or storage in a dry storage facility. The standard method to reduce the size of the Mark-18 targets would be to cut them up under water in an SRS storage basin. The condition of the Mark-18 targets presents a health and safety vulnerability for under water cutting because of the suspected brittle condition of the targets and the uncertainty concerning which portion of the target assemblies contains the americium and curium product and fission products. Because of these concerns a previous DOE assessment (see Section 1.6.2) concluded that the Department should consider processing the Mark-18 targets. Although that alternative was not chosen, and

TC

the Mark-18 targets are still stored in the Receiving Basin for Offsite Fuel, the analysis was performed and is incorporated as part of the Maximum Impact Alternative in this EIS. Processing the Mark-18 targets would not extend the operating time for the SRS canyons.

Until the Mark-51 and other Higher Actinide Targets are transferred to another site for use, DOE would continue to wet-store the material at the SRS. Additionally, DOE would continue to wet-store the non-aluminum-clad spent nuclear fuel at SRS until the material is shipped to the Idaho National Engineering and Environmental Laboratory. In the event the Mark-51 and "other" targets and non-aluminum clad fuel have not been transferred offsite by the time a dry storage facility is in operation at the SRS, DOE could repackage the targets and the fuel and transfer the material to dry storage. DOE would transfer the targets and non-aluminum clad fuel to dry storage after the material had been relocated from the Receiving Basin for Offsite Fuel to the L-Reactor Disassembly Basin in support of activities to phase out operations in the Receiving Basin for Offsite Fuel by 2007.

2.4.3 PREFERRED ALTERNATIVE

Under the preferred alternative, DOE would implement several of the technologies identified in Section 2.2 to manage spent nuclear fuel at SRS. These technologies are Melt and Dilute, Conventional Processing, and Repackage and Prepare to Ship. Each of these technologies would treat specific groups of spent nuclear fuel, as described below. The technology and fuel group combinations form DOE's Preferred Alternative in this EIS. The configuration of this preferred alternative is identified in Table 2-9. Figure 2-15 provides a flowchart for the Preferred Alternative.

2.4.3.1 Melt And Dilute

DOE has identified the Melt and Dilute process as the preferred method of treating most (about 97 percent by volume or about 32,000 MTRE) of the aluminum-based SNF considered in this EIS. DOE will continue to pursue a research and development program leading to a demon-

stration of the technology in FY 2001 using full-size irradiated research reactor spent nuclear fuel assemblies. With a successful demonstration of the technology, DOE expects to have ready a treatment facility to perform production melt and dilute operations in FY 2008. DOE will ensure the continued availability of SRS conventional processing facilities until we have successfully demonstrated implementation of the Melt and Dilute treatment technology.

The fuel proposed for the preferred Melt and Dilute technology includes the Material Test Reactor-like fuel, most of the Loose Uranium Oxide in Cans fuel, and most of the HEU/LEU Oxide and Silicide fuel. Exceptions are the failed and sectioned Oxide and Silicide fuel, about 10 percent of the Loose Uranium Oxide in Cans fuel as described in Section 2.4.3.2, and the Higher Actinide Targets and Non-Aluminum-Clad fuel that would be repackaged and prepared to ship as discussed in Section 2.4.3.3. The Melt and Dilute Technology satisfies DOE's objective and preference, as stated in the Record of Decision for the Nonproliferation Policy and Spent Nuclear Fuel EIS (60 FR 25091), to select a non-chemical separations-based technology to pre-prepare aluminum-based SNF for placement in a geologic repository. Additionally, this new technology provides significant waste reduction (of high-level, low-level, transuranic, etc.) in comparison to conventional chemical processing and is fully compatible with and supportive of the nonproliferation objectives of the United States.

The potential impacts (e.g., worker and public health, waste generation, socioeconomics, etc.) among the new non-separations based technologies were all very similar; however, the Melt and Dilute option was the most efficient in volume reduction and produced the fewest number of SNF canisters. In fact, Melt and Dilute would increase volume reduction by more than 3 to 1 over Direct Disposal/Direct Co-Disposal. The volume reduction is achieved because the melt and dilute process eliminates voids in the fuel elements and in the canisters and fuel

Table 2-9. The fuel group technology configurations that compose the preferred alternative.

Fuel group	1. Prepare for Direct Co-Disposal		2. Repackage and Prepare to Ship ^a		3. Melt and Dilute	4. Mechanical Dilution	5. Vitrification Technologies	6. Electro-metallurgical Treatment	7. Conventional Processing
A. Uranium and Thorium Metal Fuels	-	-	-	-	-	-	-	-	Preferred
B. Materials Test Reactor-Like Fuels	-	-	-	-	Preferred	-	-	-	-
C. HEU/LEU ^b Oxides and Silicides Requiring Resizing or Special Packaging	-	-	-	-	Preferred	-	-	-	Preferred ^c
D. Loose Uranium Oxide in Cans	-	-	-	-	Preferred	-	-	-	Preferred ^d
E. Higher Actinide Targets ^e	-	-	-	-	-	-	-	-	-
F. Non-Aluminum Clad Fuels	-	-	Preferred	-	-	-	-	-	-

a. This alternative describes shipment to a DOE site other than SRS, not to a geologic repository.

b. HEU = highly enriched uranium; LEU = low enriched uranium.

c. For failed or sectioned Oak Ridge Reactor fuel, High-Flux Isotope Reactor fuel, Tower Shielding Reactor fuel, Heavy Water Components Test Reactor Fuel, and a Mark-14 target (i.e., <1 percent of material in this fuel group).

d. For Sterling Forest Oxide fuel (i.e., about 10 percent of the material in this fuel group).

e. The preferred alternative is to maintain fuel Group E in continued wet storage until a decision is made on final disposition.

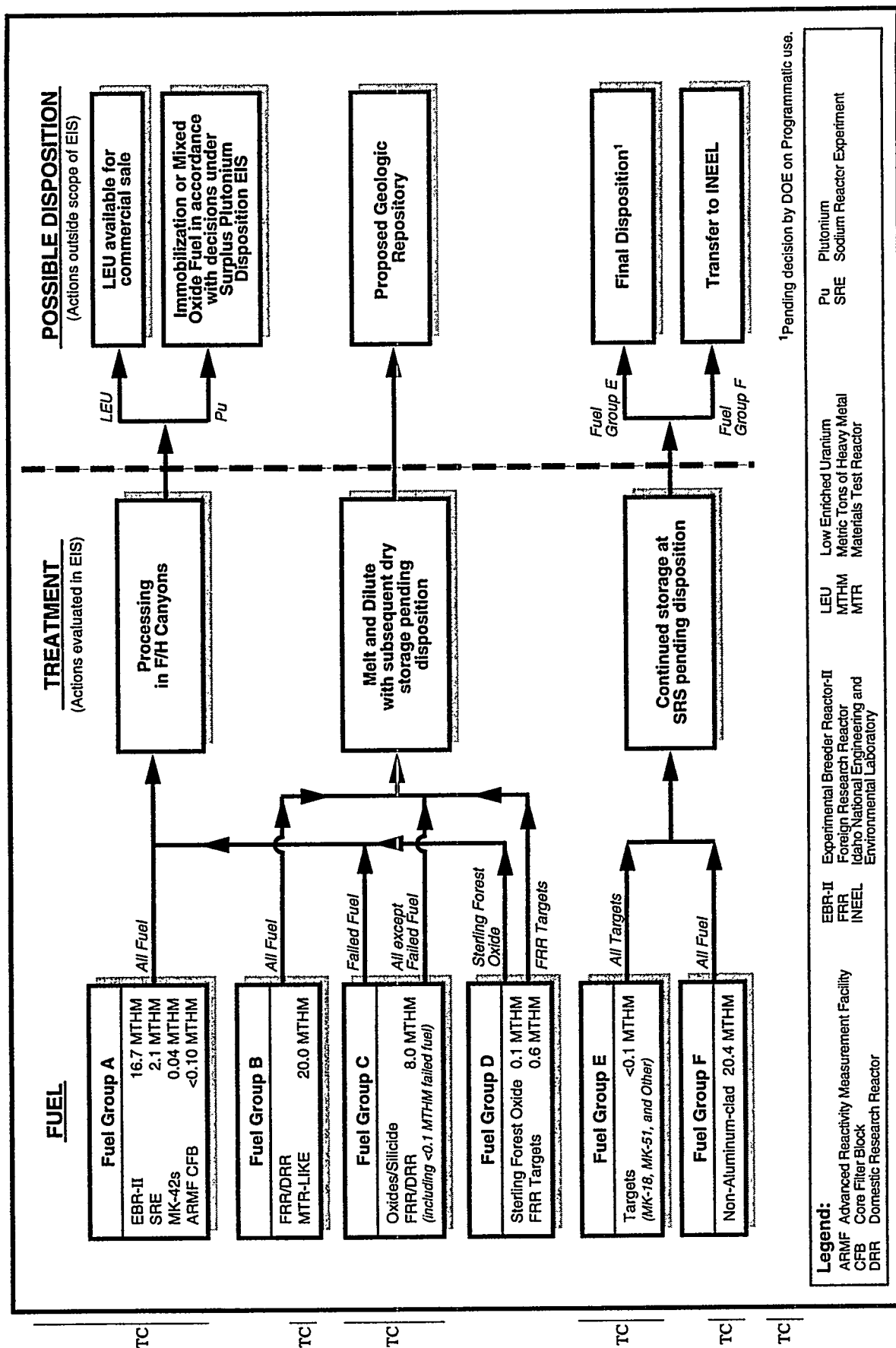


Figure 2-15. Preferred Alternative Management Flow-Path.

baskets used in the Direct Disposal/Direct Co-Disposal technology. DOE considered Melt and Dilute to be among the most "proven" of the new non-separations-based technologies because DOE has made extensive progress in the development of the melt and dilute process.

TC | The Melt and Dilute technology offers DOE the flexibility to engineer the final waste form to provide a high degree of confidence the material would be acceptable for placement in a geologic repository. Major technical concerns such as fuel characterization, criticality control, and repository performance can be reduced or eliminated by tailoring the chemical and physical form of the final product to meet specific criteria. DOE expects the Melt and Dilute option would be relatively simple to implement and would be less expensive than other similar technology options, although the ongoing technology development initiative will determine the viability of this alternative. The major technical issue for implementing this technology would be the design of an off-gas system to capture volatilized fission products. Preliminary engineering studies indicate that the system could be designed using proven approaches for managing off-gases.

EC | To implement the preferred alternative (Melt and Dilute technology), DOE would construct a melt and dilute facility in the existing 105-L building at SRS and build a dry-storage facility in L Area, near the 105-L building. DOE is proposing to use an existing facility to house the Melt and Dilute process because the existing structure can accommodate the process equipment and systems; the applicable portions of the structure will meet DOE requirements for resistance to natural hazards (e.g., earthquakes); the integral disassembly basin has sufficient capacity for all expected SNF receipts and the current Site inventory; using 105-L avoids the creation of a new radiologically controlled facility that would eventually require decontamination and decommissioning; and DOE has estimated the cost savings versus a new facility to be about \$70 million.

Using the Melt and Dilute technology, DOE would melt aluminum-based SNF and blend down any highly enriched uranium to low enriched uranium using depleted uranium that is currently stored at SRS. The material would be cast as ingots that would be loaded into stainless-steel canisters approximately 10 feet tall and 2 feet (or less) in diameter. The canisters would be placed in dry storage pending shipment to a geologic repository.

TC | During the development of the Melt and Dilute technology, DOE may determine that, for technical, regulatory, or cost reasons, the Melt and Dilute option is no longer viable. As a back-up to Melt and Dilute, DOE would continue to pursue the Direct Co-Disposal option of the New Packaging Technology and would implement this option if Melt and Dilute were no longer feasible or preferred. Direct Co-Disposal has the potential to be the least complicated of the new technologies and DOE believes this option could be implemented in the same timeframe as could the Melt and Dilute option. However, DOE believed there is greater risk in attempting to demonstrate that aluminum-based SNF, packaged according to the Direct Co-Disposal option, would be acceptable in a geologic repository. A comparison of the preferred (Melt and Dilute) and back-up (Direct Co-Disposal) technologies DOE proposes to use to manage most of the aluminum-based SNF at SRS is presented in Table 2-3.

EC | If DOE identifies any imminent health and safety concerns involving any aluminum-based SNF, DOE could use F and H Canyons to stabilize the material of concern prior to the melt and dilute facility becoming operational.

2.4.3.2 Conventional Processing

TC | DOE proposes to use conventional processing to stabilize some materials before a new treatment facility is in place. The rationale for this processing is to avoid the possibility of urgent future actions, including expensive recovery actions that would entail unnecessary radiation exposure to workers, and in one case, to manage a unique waste form (i.e., core filter block).

TC The total amount proposed for conventional processing is a relatively small volume of aluminum-based SNF at the SRS (about 3 % by volume and 40 % by mass). This material includes the Experimental Breeder Reactor-II fuel, the Sodium Reactor Experiment fuel, the Mark-42 targets and the core filter block from the Uranium and Thorium Metal fuel group; the failed or sectioned Tower Shielding Reactor, High Flux Isotope Reactor, Oak Ridge Reactor, and Heavy Water Components Test Reactor fuels and a Mark-14 target from the HEU/LEU Oxides and Silicides fuel group; and the Sterling Forest Oxide (and any other powdered/oxide fuel that may be received at SRS while H Canyon is still in operation) from the Loose Uranium Oxide in Cans fuel group. Although it is possible that a new treatment technology, such as melt and dilute, could be applied to most of these materials, DOE considers timely alleviation of the potential health and safety vulnerabilities to be the most prudent course of action because it would stabilize materials whose forms or types pose a heightened vulnerability to releasing fission products in the basin. Nonetheless, if these materials have not been stabilized before a new treatment technology becomes available, that new technology (melt and dilute) may be used rather than conventional processing.

The Experimental Breeder Reactor-II fuel and Sodium Reactor Experiment fuel are uranium metal that has been declad and stored in canisters in the Receiving Basin for Offsite Fuel. The declad fuels present a potential health and safety vulnerability. Should their existing storage containers leak, the metal fuel would corrode and release fission products to the water of the storage basin. Once the metal of the fuel is wetted, simply repackaging the fuel in a watertight container would not arrest the corrosion and, in fact, could exacerbate storage concerns since potentially explosive hydrogen gas would continue to be generated inside the storage canister as the fuel continued to corrode. An instance of water intrusion and subsequent fuel corrosion has already occurred with one Experimental Breeder Reactor-II canister stored in

the Receiving Basin for Offsite Fuel. Additionally, several problems have occurred with other uranium metal fuel in similar storage conditions at SRS (e.g., the Taiwan Research Reactor fuel with failed or missing cladding that was overpacked in canisters and stored in SRS wet basins). DOE addressed these situations by processing the failed or declad fuel in F Canyon to eliminate the health and safety vulnerability.

The failed or sectioned Tower Shielding Reactor, High Flux Isotope Reactor, Oak Ridge Reactor, and Heavy Water Components Test Reactor fuel, and a sectioned Mark-14 target from the HEU/LEU Oxides and Silicides fuel group also present potential health and safety vulnerabilities. The integrity of these fuels was destroyed for research purposes. Then the material was canned and placed in wet storage at SRS. A breach of or leak in the cans would expose the interior surfaces of the sectioned fuel to water, contaminating the water in the storage basin with radioactivity, and accelerating the corrosion of the fuel.

TC A potential health and safety vulnerability also exists for the unirradiated Mark-42 targets from the Uranium and Thorium Metal fuel group and the Sterling Forest Oxide fuel from the Loose Uranium Oxide in Cans fuel group. Should a breach occur in the cladding on the Mark-42 targets or in the canisters of Sterling Forest Oxide fuel, the particulate nature of the nuclear material in the targets and the Sterling Forest Oxide fuel could lead to dispersion of radioactive material in the water of the Receiving Basin for Offsite Fuel. Therefore, DOE is proposing to take action now to avoid the possibility of urgent future actions, including expensive recovery actions that also would entail unnecessary radiation exposure to workers.

DOE proposes to process the Experimental Breeder Reactor-II fuel and the Mark-42 targets in F Canyon. That fuel contains plutonium, approximately 114 kg of which would be recovered as part of the normal F Canyon chemical separations process and then transferred to FB-Line for conversion to metal. The plutonium

metal would be considered surplus to the nation's nuclear weapons program and would be placed in storage at the SRS pending disposition pursuant to the January 2000 Record of Decision (ROD) for the Surplus Plutonium Disposition Environmental Impact Statement (DOE 1999). The surplus plutonium would be immobilized using the can-in-canister process or fabricated into mixed-oxide (MOX) commercial power reactor fuel at the SRS. DOE has scheduled processing of the Experimental Breeder Reactor-II fuel and the Mark-42 targets in FY00.

DOE proposes to process the Sodium Reactor Experiment fuel, the failed or sectioned fuel from the HEU/LEU Oxides and Silicides fuel group, and the Sterling Forest Oxide fuel in H-Canyon where the highly enriched uranium would be blended down to low enriched uranium and stored pending potential sale as feedstock for commercial nuclear fuel. DOE would begin processing operations in H Canyon in 2000 and could complete them in about 18 months.

DOE also proposes to process the core filter block from the Uranium and Thorium Metals fuel group. The core filter block is made of depleted uranium but it contains corrosion-resistant metal (e.g., stainless-steel) that would be incompatible with the Melt and Dilute Technology for aluminum-based SNF. The core filter block could be processed in either F Canyon or H Canyon. In either case, the material would become feedstock to blend down highly enriched uranium from either conventional processing or melt and dilute operations.

The processing operations described above in both F and H Canyons would occur when the canyons were being operated to stabilize other nuclear material. It is the preference of the Department of Energy not to utilize conventional reprocessing for reasons other than safety and health. However, the core filter block is not compatible with the melt and dilute process for aluminum-based SNF. The benefit to develop a new process to accommodate this form would be disproportionately small when compared to

the cost (DOE 1998a). Consequently, the Department proposes an exception in this case.

2.4.3.3 Repackaging

DOE would continue to wet-store the non-aluminum-clad spent nuclear fuel at SRS until the material is shipped to the Idaho National Engineering and Environmental Laboratory. In the event that the non-aluminum-clad fuel has not been transferred offsite by the time a dry storage facility is in operation at the SRS (to support the Melt and Dilute Technology), DOE could repackage the fuel and transfer the material to dry storage.

2.4.3.4 Continued Wet Storage

DOE is not proposing any actions that would lead to the programmatic use of the higher actinide targets. Therefore, under the preferred alternative the Mark-18, Mark-51 and other higher actinide targets would be maintained in wet-storage until decisions are made on their final disposition.

2.4.4 DIRECT DISPOSAL ALTERNATIVE

This alternative combines the New Packaging and the New Processing Technologies with the Conventional Processing Technology. Materials Test Reactor-like fuels and HEU/LEU Oxides and Silicides (except the failed and sectioned fuels) would be treated using the Direct Disposal/Direct Co-Disposal technology and placed in the Transfer and Storage Facility with a minimum of treatment (e.g., cold-vacuum drying and canning).

DOE would manage the Higher Actinide Targets and the non-aluminum based SNF as described in the Maximum Impact Alternative.

The uranium fuel and thorium metal fuel, Sterling Forest Oxide fuel from the Loose Uranium Oxide in Cans fuel group, and failed and sectioned fuel from the HEU/LEU Oxides and Silicides fuel group would be treated using chemical separations processes under the Conventional Processing Alternative to alleviate the potential health and safety vulnerabilities dis-

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TC

TC

cussed in Section 2.4.3.2 and because this material probably would not be suitable for placement in a geologic repository if treated with the Direct Disposal/Co-Disposal option. Most of the material in the Loose Uranium Oxide in Cans fuel group would be treated using Melt and Dilute since that material could be received after a melt and dilute facility was available.

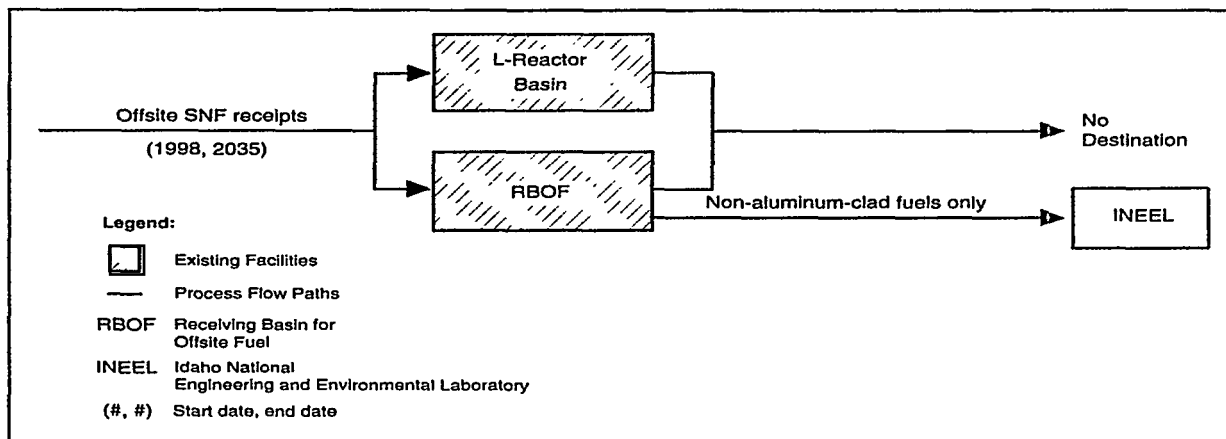
2.4.5 NO-ACTION ALTERNATIVE: CONTINUED WET STORAGE

Under the No-Action Alternative, DOE would consolidate existing inventories of SNF at SRS in the L-Reactor Disassembly Basin and the Receiving Basin for Offsite Fuel, and would store incoming SNF shipments in those basins. Maintenance, monitoring, and normal basin operations (as described in Section 2.3.1) would continue. DOE would be able to meet its commitments to receive SNF from domestic, foreign, and university research reactors and from the Idaho National Engineering and Environmental Laboratory. However, DOE would not meet the commitment made in the Record of Decision (61 FR 25092) for the Final EIS on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor

Spent Nuclear Fuel (DOE 1996c) to manage its SNF in a road-ready condition for ultimate shipment to the geologic repository. DOE could ship non-aluminum-clad fuels to the Idaho National Engineering and Environmental Laboratory in accordance with the Record of Decision (60 FR 28680) for the Programmatic SNF EIS (DOE resulting in increased environmental, safety, and health vulnerabilities. DOE would use the F or 1995b). Over the potentially 40 years of continued wet storage, some fuel could deteriorate, H Canyon facilities if they were operating for other reasons to stabilize any SNF that presented an environmental, safety, or health vulnerability. Figure 2-16 shows the No-Action Alternative.

DOE analyzed the impacts of transporting aluminum-based spent nuclear fuel to the Savannah River Site in the Nonproliferation Policy and Spent Nuclear Fuel EIS (DOE 1996c) and the programmatic SNF EIS (DOE 1995b). These documents concluded that the potential human health impacts from transportation of this fuel to SRS were low.

Figure 2-16. No-Action Alternative – Continued Wet Storage.



TC

The No-Action Alternative would be applicable to all fuel groups; however, non-aluminum-clad fuels would remain in wet storage at SRS only until DOE shipped them to the Idaho National Engineering and Environmental Laboratory in accordance with the Programmatic SNF EIS Record of Decision.

2.4.6 ALTERNATIVES NOT ANALYZED IN DETAIL

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DOE considered dry storing aluminum-based SNF (with no treatment or packaging) as a possible alternative for evaluation in this EIS. The first step for dry storing aluminum-based SNF would be accomplished by constructing a dry transfer facility. Fuel would be removed from wet storage in transfer casks, transported to the dry transfer facility, and removed from the transfer cask. Then the fuel would be placed in dry storage without any characterization, re-packaging, or treatment that would be done under the New Packaging Technology alternative or New Processing Technology alternative. DOE decided not to evaluate this alternative because it would not meet the purpose and need for agency action (i.e., it would not prepare SNF for placement in a geologic repository). In order to prepare fuel for disposition, DOE would still have to implement the New Packaging Technology, New Processing Technology, or Conventional Processing alternatives, and dry storage is already analyzed as a component of these alternatives as applicable.

DOE considered a variation to the Chemical Processing Technology option where the dissolved Experimental Breeder Reactor-II fuel would be transferred to the high-level waste tanks at the SRS for subsequent vitrification in the Defense Waste Processing Facility. DOE evaluated this action under the *Interim Management of Nuclear Materials Final Environmental Impact Statement* (DOE 1995c) for material that is very similar to the Experimental Breeder Reactor-II fuel (i.e., Mark-31 targets and Taiwan Research Reactor SNF). In that EIS, DOE concluded that the process of transferring more than trace quantities of fissile material to the high-level waste tanks with

subsequent vitrification was technically very complex and that it would take at least 6 years to develop the process. DOE noted that the Department would have to develop a process that would render fissile materials incapable of producing a nuclear criticality, regardless of the location or amount accumulated in various equipment or tanks. DOE postulated that this could be accomplished by the addition of a chemical or other material to serve as a nuclear "poison," which would minimize the potential for a criticality. However, the nuclear poison would have to be designed to accompany the fissile material throughout the process or different poisons would have to be used at different process steps (evaporation, concentration, precipitation, and ultimately vitrification). For these reasons, DOE does not consider this technology/fuel option reasonable for analysis in this EIS. Instead, DOE has analyzed the Dissolve and Vitrify option in the EIS, which would accomplish the same purpose as transferring the dissolved Experimental Breeder Reactor-II fuels to the high-level waste tanks for vitrification in the Defense Waste Processing Facility.

2.5 Comparison of Environmental Impacts Among Alternatives

Chapter 4 presents the predicted operational impacts, potential accident impacts, and construction impacts for each technology option and alternative. This organization enables the evaluation of recurring impacts (i.e., impacts from normal operations) independent of the infrequent impacts of accidents and the one-time impacts of construction.

As discussed in Section 1.3, DOE believes the amount of foreign research reactor SNF to be received in the U.S. could decrease from about 18 metric tons heavy metal (MTHM) to about 14 MTHM (or less). Therefore, the actual amount of aluminum-based material could be less than the 48 MTHM evaluated in this EIS. The only effect would be a small reduction of environmental impacts described in this EIS. DOE does not believe a reduction of this magnitude would materially affect the impacts asso-

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ciated with normal operations involving Material Test Reactor-like fuels (Fuel Group B) and the reduction would occur across all alternatives. However, where it is applicable, DOE has included information in the impact tables for normal operations that provide an example of how the reduced Fuel Group B impact data could be calculated.

The potential reduction in foreign research reactor SNF receipts would have no effect on the accident impact data that are presented in the EIS because none of the postulated accidents could affect all the fuel at once. Processing related accidents would affect only the "batch" of fuel that was involved in the process operation and accidents that could affect stored fuel, such as an earthquake, would be unlikely to involve all the fuel in the storage facility.

Impacts from normal operations under all of the alternatives would have little if any effect on ecological resources, water resources, or cultural resources. The impacts from incident-free onsite transportation of SNF would be minimal under all alternatives.

TC

Processing the Mark-18 targets (about 1 kilogram of heavy metal) was previously analyzed in the *Final Environmental Impact Statement on Interim Management of Nuclear Materials* and, therefore, was not analyzed in this EIS. The impacts of processing this small amount of material are minor and would not significantly affect the impacts analyzed for the Maximum Impact Alternative in this EIS. For example, total radiological dose from the Preferred Alternative to the maximally exposed individual for the entire period of analysis would be 0.67 millirem. Processing the Mark-18 targets would result in a dose of 0.0035 millirem.

Table 2-10 lists impacts for the five selected alternatives. The EIS identifies the following operational impacts with the potential to discriminate among the alternatives:

- *Worker and public health impacts* – Estimated impacts are reported as latent cancer

fatalities for the involved worker population, noninvolved worker, the maximally exposed member of the public, and offsite population. These impacts are summed over the period of analysis based on annual emissions and radiation doses.

Involved worker doses assume that no worker would receive more than the SRS administrative annual limit of 700 millirem. Based on this, the estimated latent cancer fatalities for the involved worker population for the entire period of analysis would range from 0.28 for the Minimum Impact Alternative to 0.84 for the Maximum Impact Alternative.

The values in Table 2-10 for health effects to the noninvolved worker, maximally exposed individual, and the offsite population for the No-Action Alternative represent current reactor-area emissions (including two SNF wet basins) for the entire period of analysis. The values for the other alternatives would be incremental above these baseline values. Summing these baseline and incremental values would be conservative, however, because there would not be two SNF wet basins operating over the entire 38-year period of analysis.

The noninvolved worker highest estimated probability of a latent cancer fatality over the entire period of analysis would range from 2.0×10^{-9} for the Minimum Impact Alternative to 6.3×10^{-7} for the Maximum Impact Alternative.

The estimated latent cancer fatality probability to the maximally exposed individual over the entire period of analysis would range from 3.0×10^{-10} (Minimum Impact Alternative) to 3.4×10^{-7} (Maximum Impact Alternative). The estimated latent cancer fatalities in the offsite population affected by SRS over the entire period of analysis would be much less than 1 for any alternative. These estimated offsite latent cancer

Table 2-10. Impact summary by alternative.

Parameter		No Action Alternative (baseline)	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative
Health Effects for the Entire Period of Analysis (1998-2035)						
TC	Latent cancer fatality probability for the noninvolved worker	$1.7 \times 10^{-6(a)}$	2.0×10^{-9}	9.6×10^{-9}	6.1×10^{-7}	6.3×10^{-7}
	Latent cancer fatality probability for the maximally exposed member of the public	$3.1 \times 10^{-7(a)}$	3.0×10^{-10}	3.6×10^{-9}	9.5×10^{-8}	3.4×10^{-7}
	Latent cancer fatalities for the worker population	0.30	0.28	0.34	0.33	0.84
	Latent cancer fatalities for the general public	$1.1 \times 10^{-2(a)}$	1.1×10^{-5}	3.8×10^{-5}	3.4×10^{-3}	4.4×10^{-3}
	Waste Generation Required for the Entire Period of Analysis (1998-2035)					
TC	Liquid (cubic meters)	2,300	660	1,200	1,050	10,500
	High-level waste generated (equivalent DWPFb canisters)	38	11	20	17	160
	Transuranic waste generated (cubic meters)	0	15	360	563	3,700
	Hazardous and mixed low-level waste generated (cubic meters)	76	25	46	103	267
	Low-level waste generated (cubic meters)	57,000	20,000	31,000	35,260	140,000
	Utilities and Energy Required for the Entire Period of Analysis (1998-2035)					
	Water consumption (millions of liters)	1,100	660	1,400	1,186	8,000
	Electricity (megawatt-hours)	46,000	27,000	81,000	116,000	600,000
	Steam (millions of kilograms)	340	190	520	650	3,600
	Diesel fuel (thousands of liters)	230	180	2,300	2,760	22,000
	Road-ready Repository canisters (1998-2035)	0	~1,400	~1,300	~400	0 ^c

a. Reflects current reactor-area emissions (including two SNF wet basins) for the entire period of analysis.

b. DWPF = Defense Waste Processing Facility.

c. The technology used in the Maximum Impact Alternative (i.e., Conventional Processing) would produce only high-level waste.

Table 2-11. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for each fuel group and technology (percent of regulatory standard).

Fuel group	Technology						
	1. Prepare for Direct Co-Disposal	2. Repackage and Prepare to Ship ^a	3. Melt and Dilute	4. Mechanical Dilution	5. Vitrification Technologies	6. Electro- metallurgical Treatment	7. Conventional Processing
A. Uranium and Thorium Metal Fuels	0.02 (ozone [as VOC])	NA	0.03 (ozone [as VOC])	No	1.1 (nitrogen ox- ides)	0.03 (ozone [as VOC])	1.1 (nitrogen ox- ides)
B. Materials Test Reactor-Like Fuels	0.03 (ozone [as VOC])	NA	0.05 (ozone [as VOC])	0.03 (ozone [as VOC])	1.7 (nitrogen ox- ides)	0.05 (ozone [as VOC])	1.7 (nitrogen ox- ides)
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	0.01 (ozone [as VOC])	NA	0.02 (ozone [as VOC])	0.01 (ozone [as VOC])	0.55 (nitrogen ox- ides)	0.02 (ozone [as VOC])	0.55 (nitrogen ox- ides)
D. Loose Uranium Oxide in Cans	NA	NA	<0.004 (ozone [as VOC])	NA	0.06 (nitrogen ox- ides)	<0.002 (ozone [as VOC])	0.06 (nitrogen ox- ides)
E. Higher Actinide Targets	NA	<0.004 (ozone [as VOC])	NA	NA	NA	NA	NA
F. Non-Aluminum-Clad Fuels	NA	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.
VOC = volatile organic compound.

Table 2-12. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for each alternative (percent of regulatory standard).

No Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative
0.03 (nitrogen oxides)	0.07 (ozone [as VOC])	1.2 (nitrogen oxides)	1.1 (nitrogen oxides)	3.6 (nitrogen oxides)

VOC = volatile organic compound.

fatalities would range from 1.1×10^{-5} to 4.4×10^{-3} .

- **Nonradiological Air Quality** – Table 2-11 presents the estimated maximum incremental concentrations of the nonradiological air pollutants that would contribute the most to the deterioration of air quality at the SRS boundary. Concentrations are presented for each technology fuel group concentration. The incremental concentrations would not affect human health. Table 2-12 presents the estimated maximum incremental concentration of the nonradiological air pollutant that would contribute the most to the deterioration of air quality at the SRS boundary for each alternative. As noted from Table 2-12, the concentration of the nonradiological constituent contributing the highest fraction of the offsite air quality standard would range from 0.03 percent of the standard for the No-Action Alternative to 3.6 percent of the standard for the Maximum Impact Alternative. Under all alternatives, nonradiological air concentrations of the SRS boundary would be well below applicable standards.
- **Waste generation** – Wastes volumes were estimated over the period of analysis. The Maximum Impact Alternative would generate the greatest volume of high-level waste, while the Minimum Impact Alternative would generate the least volume of high-level waste. For wastes generated under all alternatives, DOE would use the surplus capacity in existing SRS waste management facilities to treat, store, dispose, or recycle

the waste in accordance with applicable regulations.

- **Utilities and energy consumption** – The quantities of water, electricity, steam, and diesel fuel that would be required over the entire period of analysis were estimated.

The Maximum Impact Alternative would require the most water, electricity, steam, and diesel fuel, while the Minimum Impact Alternative would require the least. For all alternatives, water and steam would be obtained from existing onsite sources and electricity and diesel fuel would be purchased from commercial sources. These commodities are readily available and the amounts required would not have an appreciable impact on available supplies on capacities.

Accidents – DOE evaluated the impacts of potential facility accidents related to each of the alternatives. For each potential accident, the impacts were evaluated as radiation dose to the noninvolved worker, radiation dose to the offsite maximally exposed individual, collective radiation dose to the offsite population, and latent cancer fatalities to the offsite population. Table 2-13 presents the results of this analysis. Table 2-13 also indicates the estimated frequency of occurrence for each accident.

The highest consequence accident postulated under the continued wet storage, direct co-disposal, and repackaging and prepare to ship technologies is a seismic/high wind-induced criticality, which is estimated to

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Table 2-13. Estimated maximum consequence accident for each technology.

		Consequences			
Option	Accident Frequency	Noninvolved Worker (rem)	MEI (rem)	Offsite Population (person-rem)	Latent Cancer Fatalities
Continued Wet Storage (No Action)^a					
RBOF (high wind-induced criticality)	Once in 26,000 years	13	0.22	12,000	6.2
L-Reactor basin (basin-water draindown)	Once in 500 years	0.014	0.016	(b)	(b)
Direct Co-Disposal					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Repackage and Prepare to Ship					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Conventional Processing					
Processing phase in F/H Canyons (coil and tube failure)	Once in 14,000 years	13	1.3	78,000	39
Melt and Dilute					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Melt and dilute phase (earthquake induced spill with loss of ventilation)	Once in 200,000 years	30	0.5	21,000	10
Mechanical Dilution					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Mechanical dilution phase (criticality with loss of ventilation)	Once in 33,000 years	0.71	0.074	3,000	1.5
Vitrification Technologies					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Vitrification phase (earthquake-induced release with loss of ventilation)	Once in 200,000 years	0.10	0.0017	71	0.035
Electrometallurgical Treatment					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Electrometallurgical phase (metal melter earthquake induced spill with loss of ventilation)	Once in 200,000 years	30	0.5	21,000	10

MEI = Maximally Exposed Individual.

RBOF = Receiving Basin for Offsite Fuels.

a. All alternatives would use RBOF and the L-Reactor Disassembly Basin; therefore, accidents in these facilities are possible for each technology.

b. Not available.

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result in 6.2 latent cancer fatalities in the offsite population. The highest consequence accident under conventional processing technology is a coil and tube failure with an estimated offsite population impact of 39 latent cancer fatalities. The frequencies of these accidents are once in 2,000 to once in 26,000 years.

For the other new SNF technologies evaluated, the maximum consequence accident (earthquake induced spill with loss of ventilation) is associated with the melt and dilute process. This accident is estimated to occur once in 200,000 years and to result in 10 latent cancer fatalities in the offsite population.

Construction activities could affect four parameters: surface-water quality, air quality, ecological resources, and socioeconomics. However, because current SRS construction workers would build the facilities in an existing industrialized area of the Site, DOE expects little impact from construction activities.

2.6 Other Decisionmaking Factors

2.6.1 TECHNOLOGY AVAILABILITY AND TECHNICAL FEASIBILITY

The New Packaging and New Processing Technology Alternatives would rely on technologies that have not been applied to the management of aluminum-based SNF for ultimate disposition. Therefore, DOE conducted a feasibility study of the non-processing technologies and documented the study in a report prepared by a Research Reactor Task Team in its Office of Spent Fuel Management (DOE 1996b).

The Research Reactor Task Team examined a wide range of technical issues involved in achieving safe and cost-effective disposal of aluminum-based SNF under DOE jurisdiction. The Team identified and evaluated issues on technical grounds to arrive at a recommended-course of action that could lead to the implementation of a non-processing SNF management technology by 2000. The team considered three

specific areas of investigation to be key: (1) repository and waste form considerations; (2) SNF receipt, handling, and storage provisions; and (3) treatment technologies (the same technologies this EIS considers). The team assigned the highest confidence of success and greatest technical suitability to technologies that would have relatively simple approaches (i.e., Direct Disposal, Direct Co-Disposal, Melt and Dilute, and Press and Dilute). The Conventional Processing option would have the least technical uncertainty because it would rely largely on a technology that is proven for aluminum-based SNF. The No-Action Alternative would involve the greatest technical uncertainty in the area of potential fuel degradation, as a result of continued long-term wet storage in SRS basins. The non-processing technologies with the greatest technical uncertainties would be the more complicated technologies such as vitrification.

In response to a DOE request, the National Academy of Sciences evaluated and provided recommendations for DOE's aluminum-based SNF disposition technical program (NAS 1998). The NAS report was prepared by a Principal Investigator assisted by a panel of expert consultants in fields of nuclear criticality control, proliferation policy, costs and schedules, corrosion and metallurgy, processing and remote handling, and regulatory waste acceptance.

The panel reviewed the DOE program for developing a strategy for treatment of aluminum-based SNF in preparation for interim storage and final disposal, with emphasis on the following objectives:

- Evaluation of the set of technologies proposed by DOE for aluminum-based SNF treatment, with suggestions of other applicable technologies
- Examination of waste package performance criteria developed by DOE to meet the anticipated waste acceptance criteria for storage, transportation, and repository disposal

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- Assessment of projected costs and schedule for implementation of the aluminum-based SNF technologies

The NAS report generally endorsed the projected DOE spent fuel disposition scenarios under development. The NAS recommendations for systems approach and phased strategy were incorporated by DOE into the EIS as follows:

Two systems analyses were completed for the primary new technologies being considered by DOE (Melt and Dilute and Direct Disposal/Direct Co-Disposal). A variety of attributes were evaluated, including cost, criticality concerns, public safety, worker safety, environmental concerns, nonproliferation, versatility, maintainability, and repository volume. One analysis was performed by Westinghouse Savannah River Company (WSRC 1998b), and a second independent multi-attribute decision analysis was completed by Sandia National Laboratory (SNL 1998). In both studies, Melt and Dilute had the least uncertainty.

DOE has recognized the advantages of applying a phased strategy for implementation of the melt and dilute process and continues to integrate its development and installation with other site program priorities and schedules in mind. The NAS concern regarding technology selection being driven by post-2015 SNF receipts is mitigated by the plan to design a facility with minimal-sized processing capabilities, which will be able to treat the current inventory of spent nuclear fuels within a reasonable timeframe, yet not be operationally burdensome when fuel receipts are reduced to minimal amounts.

The phased strategy was accommodated by provisions of backup treatments for appropriate fuel types should the projected preferred treatments not be successfully implemented within required time constraints. For example, the Direct Disposal/Direct Co-disposal technology is included as a backup technology for Melt and Dilute technology.

In summary conclusions, the NAS noted the complexity of the aluminum-based SNF disposal

program including factors such as: the timely provision of initial storage capacity for the fuel at SRS; the selection, development, and implementation of one or more treatment options to qualify the fuel for possible repository disposal; and the interim storage required until the repository, yet-to-be designed, licensed, or constructed, can accept it. The Academy noted that an SNF disposition program requires a systems approach for optimization of the many interacting factors required for successful implementation. The NAS recommended that aluminum-based SNF treatment decisions be made using a phased strategy in which critical decisions are made as the information needed for sound choices becomes available, recognizing the trade-offs between information acquisition and costs of delayed decisions.

The NAS panel identified a number of specific findings with recommendations as described in their report (NAS 1998).

Specific observations of the panel included the following:

- DOE has identified a reasonably complete set of aluminum-based SNF treatment options, resulting in selection of the Direct Co-Disposal and Melt and Dilute technologies for further development.
- The selection of a preferred treatment alternative must take into account uncertainties in repository Waste Acceptance Criteria that could, for example, disqualify highly enriched uranium waste forms such as produced by the Direct Co-Disposal technology.
- Both the Direct Disposal/Direct Co-Disposal and Melt and Dilute technologies apparently can be implemented to produce acceptable waste forms. The high-temperature Melt and Dilute treatment is technically more demanding than the relatively straight-forward Direct Disposal/ Direct Co-Disposal treatment and presents potential problems in radioactive off-gas control, but the basic operations have been

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demonstrated in other programs. Suitability of other technology options, such as the Electrometallurgical Treatment, is less assured because of the additional development work needed.

- More careful consideration of the conventional processing option is needed, because it is a well-demonstrated technology, its costs and risks are known, the necessary facilities are in current operations, and the high-level waste form is likely acceptable in the repository.
- DOE has established a working relationship with DOE-Yucca Mountain and plans to continue this relationship to ensure timely identification of repository waste form criteria and waste characterization requirements.
- Other waste form criteria, including interim-storage criteria, appear reasonable and complete, except for transportation requirements. The panel recommended DOE review shipping requirements before finalization of canister/shipping cask design for the waste forms.
- Work under way by DOE-SR appears properly focused and appropriate to the above requirements. However, a single treatment option may not be suitable for all types of aluminum-clad SNF and the program should maintain flexibility in technology selection to accommodate this variability.
- Major cost factors are accounted for in the cost projections, but schedule projections appear ambitious, and schedule delays could affect the cost projections. Projected costs are, however, not a major discriminator of the various treatments and treatment selection can proceed based on current projections.

The DOE-SR and the Nuclear Regulatory Commission (NRC) have established an agreement for the NRC to provide technical assistance in connection with the identification of potential issues relating to the placement of

aluminum-based foreign and domestic research reactor spent nuclear fuel in a geologic repository. In a recent review of DOE's research and development work, the NRC staff indicated that both the Melt and Dilute and Direct Co-Disposal technologies would be acceptable concepts for the disposal of aluminum-based research reactor SNF in a repository (Knapp 1998).

2.6.2 NONPROLIFERATION, SAFEGUARDS AND SECURITY

On May 13, 1996, the United States established a new 10-year policy to accept and manage foreign research reactor spent nuclear fuel containing uranium enriched in the United States (61 FR 25091). The goal of this policy is to reduce civilian commerce in weapons-usable highly enriched uranium, thereby reducing the risk of nuclear weapons proliferation, as called for in President William Clinton's September 27, 1993, Nonproliferation and Export Control Policy.

Two key disposition options under consideration for managing SNF in this EIS include conventional processing and new treatment and packaging technologies. The Record of Decision for managing foreign research reactor SNF specified that, while evaluating the processing option, "DOE will commission or conduct an independent study of the nonproliferation and other (e.g., cost and timing) implications of chemical separation of spent nuclear fuel from foreign research reactors." DOE's Office of Arms Control and Nonproliferation conducted the study. To receive a copy, contact DOE at 1-800-881-7292.

The study addresses the nonproliferation implications the Department considered in determining how to manage aluminum-based SNF at the Savannah River Site, including how to place these materials in forms suitable for ultimate disposition (DOE 1998a). Because the same technology options are being considered for the foreign research reactor and the other aluminum-based spent nuclear fuels, the report addresses the nonproliferation implications of

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managing all the Savannah River Site aluminum-based SNF.

The nonproliferation assessment evaluates the extent to which each technology option supports the United States nonproliferation goals, which are summarized below.

- To reduce the risk of nuclear proliferation and for other considerations, the United States neither encourages the civil use of plutonium nor engages in plutonium processing for either nuclear power or nuclear explosive purposes. In addition, the United States works actively with other nations to reduce global stocks of excess weapons-usable material; separated plutonium and highly enriched uranium. Under this policy, the United States honors its commitments to cooperate with civilian nuclear programs that involve the processing and recycling of plutonium in Western Europe and Japan. In all such cases, however, the United States seeks to ensure that the International Atomic Energy Agency (IAEA) has the resources needed to implement its vital safeguards responsibilities, and works to strengthen the IAEA's ability to detect clandestine nuclear activities. The United States seeks to eliminate where possible the accumulation of stockpiles of highly enriched uranium or plutonium, and to ensure that where these materials already exist they are subject to the highest standards of safety, security, and international accountability. The United States also actively opposes, as do other supplier nations, the introduction of processing and plutonium recycling activities in regions of proliferation concern.
- The United States also seeks to minimize the adverse environmental, safety, and health impacts of its management of nuclear materials and activities. This goal includes minimizing the generation of radioactive wastes and ensuring that waste materials are put into forms that can be disposed of safely.

To evaluate the extent to which the technology options support the United States' nonproliferation policy goals, the nonproliferation study evaluated the technology options using technical and policy factors, as explained below.

Technical factors include the degree to which a particular technology would:

- Help ensure that the weapons-usable nuclear material in the spent nuclear fuel could not be stolen or diverted during the process. This includes an assessment of the attractiveness to diversion of materials in process and the ease of providing institutional and inherent security features.
- Facilitate cost-effective international verification and transparency.
- Result in converting the spent nuclear fuel into a form from which retrieval of the material for weapons use would be difficult and unlikely, thus modestly reducing the total stockpile of material readily usable in nuclear weapons.

Policy factors include the degree to which a particular technology would:

- Be consistent with United States policy related to processing and nonproliferation.
- Avoid encouraging other countries to engage in the processing of spent nuclear fuel, or undermining United States efforts to limit the spread of processing technology and activities, particularly to regions of proliferation concern.
- Support United States efforts to convert United States and foreign research reactors to low enriched fuels, and avoid creating technical, economic, or political obstacles to implementing the Foreign Research Reactor Spent Nuclear Fuel Acceptance Program.
- Help demonstrate that any treatment of these spent nuclear fuels will definitely not represent the production by the United

States of additional materials for use in nuclear weapons.

- Support negotiation of a nondiscriminatory global fissile material cutoff treaty.

There are several options for the effective management of the aluminum-based SNF at SRS.

With respect to nonproliferation, the report concluded the following:

- All of the options could reliably discourage any theft or diversion of the material, but some are superior to others.
- All of the options could provide for some form of international safeguarding by the International Atomic Energy Agency (IAEA). The options vary in terms of cost and ease of application.
- All of the options would result in forms from which recovery of the material for use in weapons would be highly unlikely, although the Direct Disposal/Direct Co-Disposal Option would not blend down the residual highly enriched uranium and low enriched uranium, and the conventional processing option would recover plutonium metal that would be managed as surplus.
- All of the options would be consistent with United States nonproliferation policy, and would allow for verification approaches that would be acceptable to the United States if implemented in other countries.
- The electrometallurgical treatment and the conventional processing, by appearing to endorse these technologies, could conceivably encourage processing in other countries.
- All of the options have the potential to support fully United States efforts to reduce the civil use of highly enriched uranium, including the Foreign Research Reactor Spent Nuclear Fuel Acceptance Program.
- None of these options would appear to be prejudicial to the ability of the United States to submit to international safeguards or monitoring under a nondiscriminatory fissile material cutoff treaty. However, the processing option involves the use of old facilities at the Savannah River Site not specifically designed to facilitate the application of international safeguards. An effective safeguarding regime would likely be difficult due to cost and safety retrofitting concerns (DOE 1998a).
- The Office of Arms Control and Nonproliferation fully supports the active pursuit of a new treatment technology for the aluminum-based spent nuclear fuel, and views the melt and dilute recommendation as a favorable technology in light of nonproliferation concerns.

2.6.3 LABOR AVAILABILITY AND CORE COMPETENCY

Each alternative and associated technologies would require different levels of personnel knowledge and training. In addition, providing the needed level of training would result in impacts, primarily in the area of personnel resources. In general, the New Packaging options probably would be the least labor-intensive. The Conventional Processing option or a combination of options that included conventional processing would be the most labor-intensive to implement on an annual basis.

Operations required for the Conventional Processing technology would occur in parallel with other canyon nuclear stabilization programs. As a result, no excess personnel would be available in the event the vulnerable SNF was not processed. Because the canyons already would be operating to process materials not considered in this EIS, there also would be no actual cost savings that could be transferred to another activity.

The Conventional Processing technology option and No-Action Alternative would require the least amount of training because the SRS

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workforce has a great deal of experience in these technologies and there are existing training and qualification programs to maintain core competency. The New Processing Technology options such as Vitrification Technologies or Electrometallurgical Treatment probably would require the greatest training effort because they would involve new and complex operations.

2.6.4 MINIMUM CUSTODIAL CARE

The New Packaging Technology and New Processing Technology options would create a form of material that required the least amount of custodial care before shipment off the Site. However, safeguards and security requirements would still be maintained. Conventional processing would require care of the vitrified waste similar in level-of-effort to the custodial care of the New Packaging and New Processing Technology option. In addition, it also would require care of the high-level waste until it was vitrified and any blended-down fissile material until they were delivered for disposition.

2.6.5 COST

To determine the potential cost of integrating various combinations of alternatives, DOE has estimated life-cycle costs for the alternatives and for the new technology options described in this EIS and for conventional processing. The cost report was prepared, in part, to satisfy the Department's commitment to study the implications of chemically separating SNF (see Section 2.6.2). The planning level costs have an uncertainty of +50 percent to -30 percent. These estimates, which are listed in Table 2-14,

include both operating and capital (i.e., construction) costs (DOE 1998b).

DOE estimated the costs for the alternatives discussed in this EIS using the technology option cost information from the cost study. The cost estimates for the alternatives are presented in Table 2-15.

Comparison of the projected life cycle costs for the alternatives indicate the following:

- The life-cycle costs range from a low of \$1.7 billion for No Action to a high of \$2.0 billion for the Maximum Impact Alternative. However, the continued wet storage cost does not include actions necessary to prepare SNF for ultimate disposition.
- The Direct Disposal Alternative (\$1.9 billion) and the Preferred Alternative (\$2.0 billion) (both using a renovated reactor building) have approximately the same life-cycle cost, with installation in a renovated reactor facility presenting cost advantages of about \$200 million compared to a new treatment facility.
- The cost of processing the SNF proposed in the Preferred Alternative would be incremental to the cost of operating the canyons for other reasons and very small when compared to the canyon overall operating cost.

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Table 2-14. Life-cycle costs for aluminum-clad fuel technologies (1998 millions of dollars).^a

		Technology								
1	1A	2	3	3A	4	5	6	7	8	
Cost factors	Prepare for Direct Co-Disposal	Prepare for Direct Co-Disposal in a Renovated Reactor Building	Repackage and Prepare to Ship ^b	Melt and Dilute	Melt and Dilute in a Renovated Reactor Building	Mechanical Dilution	Vitrification Technologies	Electro-metallurgical Treatment	Conventional Processing ^c	Continued Wet Storage
Wet storage and handling	676	766	NA	676	766	676	676	676	655	1,650
Transfer, storage, and treatment	1,241	919	NA	1,363	1,073	1,566	2,411 ^d	2,625	765	0
Fuel and waste processing	33	37	NA	47	55	46	67	67	610	78
Repository disposal ^e	169	169	NA	56	56	82	198 ^f	23	36	0
Total	2,119	1,891	NA	2,142	1,950	2,370	3,352 ^g	3,391	2,066 ^h	1,730

EC	a.	DOE (1998b).
	b.	Repackage and Prepare to Ship activities would be accomplished under Prepare for Direct Co-Disposal. The costs are included in those reported for Prepare for Co-Disposal. The material would be shipped to another DOE site. NA = not applicable.
L14-4	c.	Value is for Conventional Processing of SNF until FY 2010, followed by Melt and Dilute treatment of later fuel receipts. This reduces the size of the facility for the later fuel receipts and lowers the overall life-cycle cost of the Conventional Processing technology by about \$400 million.
	d.	Value is for the Dissolve and Vitrify technology. Glass Material Oxidation and Dissolution System would be 2,145 and Plasma Arc Treatment would be 2,143.
	e.	Costs for shipping the final waste form from SRS to a repository are included in these cost projections.
	f.	Value is for the Dissolve and Vitrify and Glass Material Oxidation and Dissolutions System technologies. Plasma Arc Treatment would be 80.
	g.	Value is for the Dissolve and Vitrify technology. Glass Material Oxidation and Dissolution System would be 3,087 and Plasma Arc Treatment would be 2,966.
L5-16, L14-3	h.	Credits for sale of recovered enriched uranium are not included because of the recently signed agreement between Russia and the U.S. that calls for potential deferment of enriched uranium sales. Including these credits would decrease the cost of the electrometallurgical treatment by about \$150 million and the Conventional Processing by about \$110 million.

EC a. DOE (1998b).

b. Repackage and Prepare to Ship activities would be accomplished under Prepare for Direct Co-Disposal. The costs are included in those reported for Prepare for Co-Disposal. The material would be shipped to another DOE site. NA = not applicable.

c. Value is for Conventional Processing of SNF until FY 2010, followed by Melt and Dilute treatment of later fuel receipts. This reduces the size of the facility for the later fuel receipts and lowers the overall life-cycle cost of the Conventional Processing technology by about \$400 million.

d. Value is for the Dissolve and Vitrify technology. Glass Material Oxidation and Dissolution System would be 2,145 and Plasma Arc Treatment would be 2,143.

e. Costs for shipping the final waste form from SRS to a repository are included in these cost projections.

f. Value is for the Dissolve and Vitrify technology. Glass Material Oxidation and Dissolutions System technologies. Plasma Arc Treatment would be 80.

g. Value is for the Dissolve and Vitrify technology. Glass Material Oxidation and Dissolution System would be 3,087 and Plasma Arc Treatment would be 2,966.

h. Credits for sale of recovered enriched uranium are not included because of the recently signed agreement between Russia and the U.S. that calls for potential deferment of enriched uranium sales. Including these credits would decrease the cost of the electrometallurgical treatment by about \$150 million and the Conventional Processing by about \$110 million.

Table 2-15. Life-cycle costs (1998 billions of dollars) for each alternative.^a

Minimum Impact	Direct Disposal	Preferred Alternative	Maximum Impact	No Action
1.9 ^b	1.9	2.0 ^c	2.0	1.7

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a. Source: DOE (1998b).

b. Includes less than \$30 million to install Melt and Dilute capability for Fuel Group D.

c. Includes about \$6 million as direct and indirect cost of operating canyons for SNF processing during 1999-2001 while the material stabilization program is underway in response to Defense Nuclear Facility Safety Board Recommendation 94-1.

References

- Bugher, W. C., 1998, "Life-Cycle Cost Analysis of the 105-L SNF TSS Project (U)," 400:ECD-SEG-98-0002, memorandum to M. Barlow, Westinghouse Savannah River Company, Aiken, South Carolina.
- Burnfield, D., 1995, "Savannah River Site Spent Fuel Storage Trip Report, January 3-6, 1995," memorandum to G. W. Cunningham, Defense Nuclear Facilities Safety Board, Washington, D.C., January 23.
- Conway, J. T., 1996, letter to The Honorable Alvin L. Alm, (U.S. Department of Energy), Defense Nuclear Facilities Safety Board, Washington, D.C., November 6.
- DNFSB (Defense Nuclear Facilities Safety Board), 1994, "Recommendation 94-1 to the Secretary of Energy pursuant to 42 U.S.C. 2286a(5) Atomic Energy Act of 1954, as amended," Washington, D.C., May 26.
- DOE (U.S. Department of Energy), 1993, *Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear Fuel and other Reactor Irradiated Nuclear Materials and their Environmental, Safety and Health Vulnerabilities*, Volume I, Washington, D.C.
- DOE (U.S. Department of Energy), 1995a, *Preliminary Requirements for the Disposition of DOE Spent Nuclear Fuel in a Deep Geologic Repository*, Washington, D.C.
- DOE (U.S. Department of Energy), 1995b, *Final Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement*, DOE/EIS-0203, Idaho Operations Office, Idaho Falls, Idaho.
- DOE (U.S. Department of Energy), 1995c, *Final Environmental Impact Statement, Interim Management of Nuclear Materials*, DOE-EIS-0220, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1995d, *Analysis of the Total System Life Cycle Cost of the Civilian Radioactive Waste Management Program*, DOE/RW-0479, Washington, D.C., September.
- DOE (U.S. Department of Energy), 1996a, *Waste Acceptance System Requirements Document*, WA-SRD, Office of Civilian Radioactive Waste Management, Washington, D.C.
- DOE (U.S. Department of Energy), 1996b, *Technical Strategy for the Treatment, Packaging, and Disposal of Aluminum-Based Spent Nuclear Fuel, A Report of the Research Reactor Spent Nuclear Fuel Task Team*, Volume 1, Washington, D.C.
- DOE (U.S. Department of Energy), 1996c, *Final Environmental Impact Statement, Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel*, DOE/EIS-0218F, Assistant Secretary for Environmental Management, Washington, D.C.
- DOE (U.S. Department of Energy), 1998a, *Nonproliferation Impact Assessment for the Management of the Savannah River Site Aluminum-Based Spent Nuclear Fuel*, Office of Arms Control and Nonproliferation, Washington, D.C., December.

EC

- DOE (U.S. Department of Energy), 1998b, *Report on the Savannah River Site Aluminum-Clad Spent Nuclear Fuel Alternative Cost Study*, Savannah River Operations Office, Aiken, South Carolina, December. EC
- DOE (U.S. Department of Energy), 1999, *Surplus Plutonium Disposition Final Environmental Impact Statement*, DOE/EIS-0283, Office of Fissile Material Disposition, Washington, D.C. EC
- Knapp, M. R., 1998, "Letter to John Anderson, Subject: Review of the Aluminum-Based Research Spent Nuclear Fuel Disposition Program," U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, Washington, D.C., June 5. EC
- Knapp, M. R., 1998, "Review of the Aluminum-based Research Reactor Spent Nuclear Fuel Disposition Program," letter from Nuclear Regulatory Office Commission of Nuclear Material Safety and Safeguards to J.E. Anderson, Acting Assistant Manager for Material and Facility Stabilization, U.S. Department of Energy, Savannah River Operations Office, Aiken, South Carolina. EC
- NAS (National Academy of Sciences), 1998, *Research Reactor Aluminum Spent Fuel – Treatment Options for Disposal*, M. Levenson (Principal Investigator), Board on Radioactive Waste Management, National Academy Press, Washington, D.C.
- SNL (Sandia National Laboratory), 1998, "A Multi-Attribute Utility Decision Analysis for Treatment Alternatives for the DOE-SR Aluminum-Based Spent Nuclear Fuel," Sandia National Laboratory Report, SAND98-2146, October. EC
- Wike, L. D., J. B. Gladden, and W. S. Large, 1996, *Preliminary Site Selection for the Spent Nuclear Fuel Transfer and Storage Facility*, WSRC-TR-0398, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- WSRC (Westinghouse Savannah River Company), 1998a, *Disposability Assessment: Aluminum-Based Spent Nuclear Fuel Forms*, WSRC-TR-98-00227, Aiken, South Carolina.
- WSRC (Westinghouse Savannah River Company), 1998b, "Spent Nuclear Fuel Alternative Technology Decision Analysis (U)," WSRC-U-ESR-G-00004, Revision 0, June 26. EC
- 55 FR 22520 (Volume 55 *Federal Register* 22520), 1990, "Land Disposal Restrictions for Third Scheduled Wastes," Volume 55, Number 105, U.S. Environmental Protection Agency, Washington, D.C., pp. 22520-22720, June 1.
- 60 FR 28680 (Volume 60 *Federal Register* page 28680), 1995, "Record of Decision for the Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs," Volume 60, Number 105, U.S. Department of Energy, Washington D.C. pp. 28680-28696, June 1.
- 60 FR 65300 (Volume 60 *Federal Register* page 65300), 1995, "Record of Decision for the Interim Management of Nuclear Materials at the Savannah River Site," Volume 60, Number 243, U.S. Department of Energy, Washington, D.C., pp. 65300-65316, December 19.
- 61 FR 25091 (Volume 61 *Federal Register* page 25091), 1996, "Record of Decision for the Final Environmental Impact Statement for Proposed Nuclear Weapons Non Proliferation Policy Concerns

for Foreign Research Reactor Spent Nuclear Fuel, DOE/EIS-218F," Volume 61, Number 97, U.S. Department of Energy, Washington D.C., pp. 25091-25103, March 17.

65 FR 1608 (Volume 65 Federal Register page 1608), 2000, "Record of Decision for the Surplus Plutonium Disposition Final Environmental Impact Statement," Volume 65, Number 7, U.S. Department of Energy, Washington, D.C., pp. 1608-1620. | EC

CHAPTER 3. AFFECTED ENVIRONMENT

This chapter describes the existing environmental and socioeconomic characteristics of the Savannah River Site (SRS) and the nearby region that the proposed action or its alternatives (described in Chapter 2) could affect. It provides the environmental bases against which the U.S. Department of Energy (DOE) has assessed the environmental consequences described in Chapter 4.

The activities that DOE describes in this environmental impact statement (EIS) would occur on the SRS, primarily in industrialized areas (for example see Figure 2-13). The only exception would involve the transportation of spent nuclear fuel or waste between SRS areas.

The industrialized areas consist primarily of buildings, paved parking lots, and graveled areas. There are grassed areas around some buildings, and there is vegetation along drainage ditches, but most of the industrialized areas have little or no vegetation.

As discussed in Section 2.4.2, DOE has identified three candidate host sites for the potential construction of a Transfer and Storage Facility. These sites are the east side of L Area inside the facility fence (see Figure 2-8), the southeast side of C Area inside the facility fence (see Figure 2-13), and the northeast side of P Area (see Figure 2-14). DOE also could construct a new Transfer, Storage and Treatment Facility at any of these three sites or in F or H Area. Finally, facilities to implement the New Processing Technology options could be located inside a reactor building, such as Building 105-L.

3.1 Geologic Setting and Seismicity

The SRS is in west-central South Carolina, approximately 100 miles from the Atlantic coast (Figure 3.1-1). It is on the Aiken Plateau of the Upper Atlantic Coastal Plain about 25 miles (40 kilometers) southeast of the Fall Line which

separates the Atlantic Coastal Plain from the Piedmont.

3.1.1 GENERAL GEOLOGY

In South Carolina, the Atlantic Coastal Plain Province consists of a wedge of seaward-dipping and thickening unconsolidated and semiconsolidated sediments that extend from the Fall Line to the Continental Shelf (Figure 3.1-1). The Aiken Plateau is the subdivision of the Coastal Plain that includes the location of the SRS. The plateau extends from the Fall Line to the oldest of several scarps incised in the Coastal Plain sediment. The Plateau surface is highly dissected and characterized by broad interfluvial areas with narrow steep-sided valleys. It is generally well drained, although poorly drained depressions (called Carolina bays) occur (DOE 1995a). At the Site, the plateau is underlain by 500 to 1,400 feet (150 to 420 meters) of sands, clays, and limestones of Tertiary and Cretaceous age. These sediments are underlain, in turn, by sandstones of Triassic age and older metamorphic and igneous rocks (Arnett and Mamatey 1996). Because of the proximity of the SRS to the Piedmont Province, it has more relief than areas that are nearer the coast, with onsite elevations ranging from 89 to 420 feet (27 to 128 meters) above mean sea level.

The sediments of the Atlantic Coastal Plain (Figure 3.1-2) dip gently seaward from the Fall Line and range in age from Late Cretaceous to Recent. The sedimentary sequence thickens from essentially 0 at the Fall Line to more than 4,000 feet (1,219 meters) at the coast. Regional dip is to the southeast. Coastal Plain sediments underlying the SRS consist of sandy clays and clayey sands, although occasional beds of clean sand, gravel, clay, or carbonate occur (DOE 1995a). The formations of interest in C, F, H, L, and P Areas are part of the shallow (Floridan) aquifer system (Figure 3.1-2 and Table 3.1-1). Any contaminants could migrate to these formations and be carried by them to SRS streams.

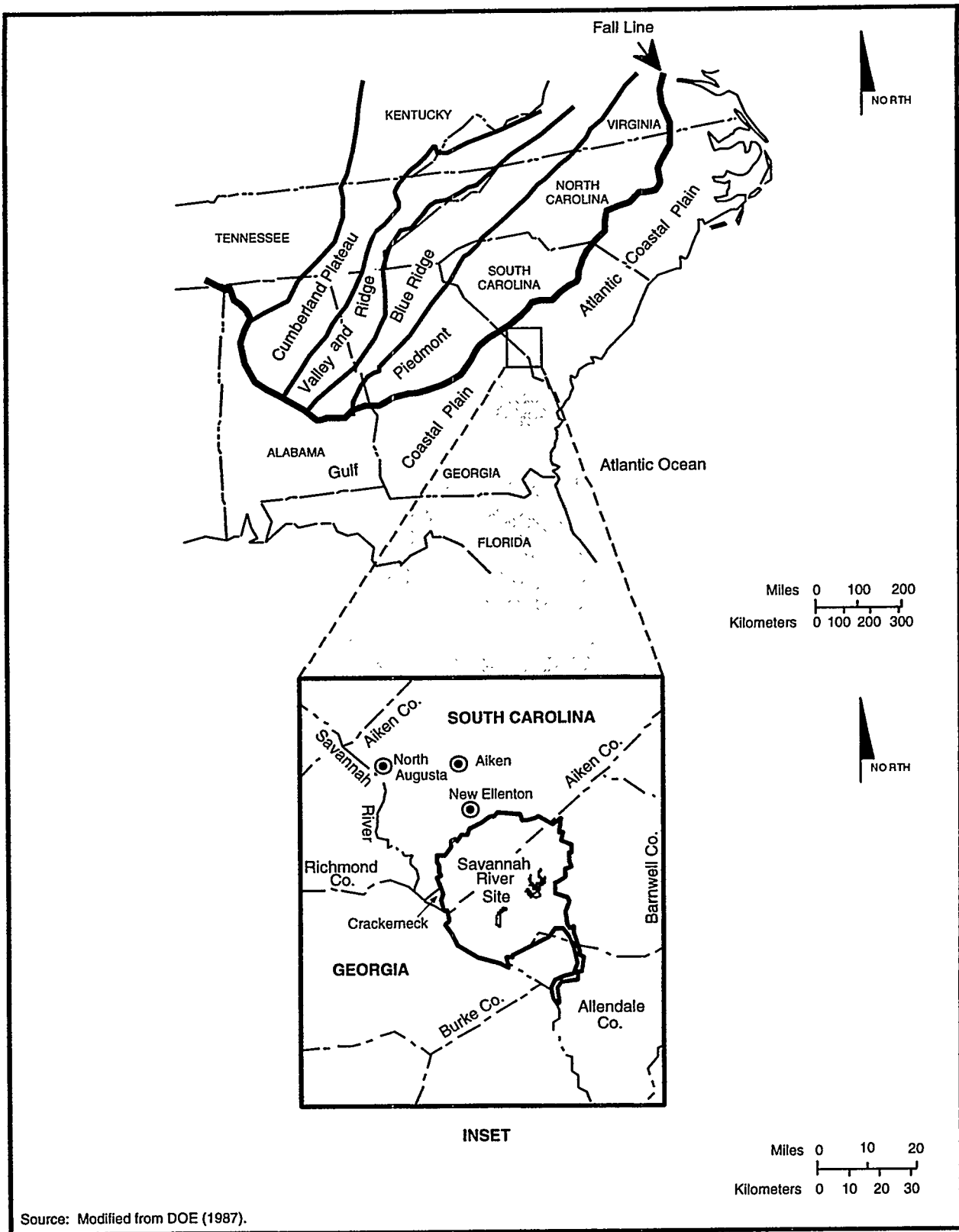
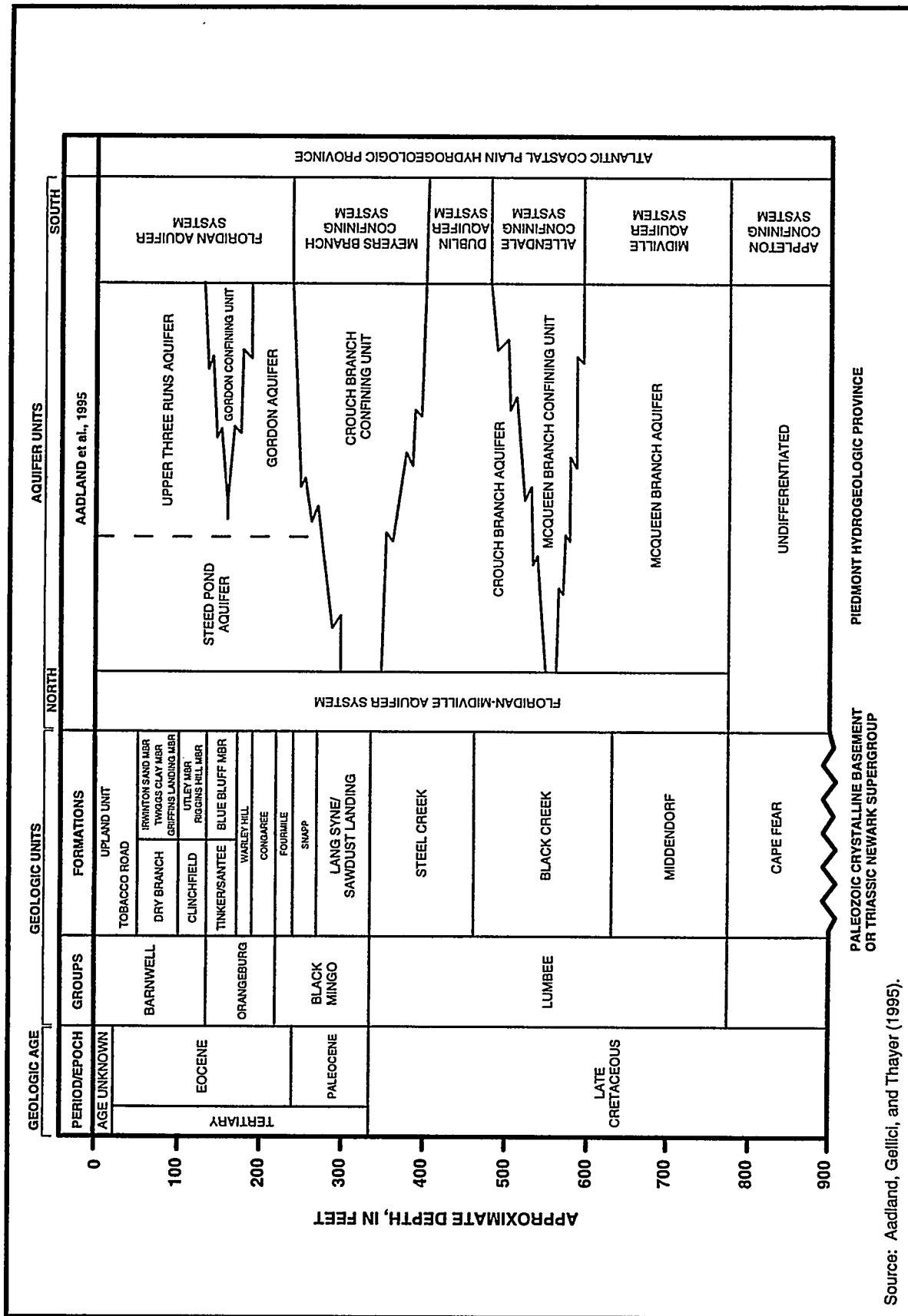


Figure 3.1-1. Generalized location of Savannah River Site and its relationship to physiographic provinces of southeastern United States.



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Figure 3.1-2. Generalized geologic and aquifer units in SRS region.

Table 3.1-1. Soil formations of the Floridan aquifer system.^a

Aquifer Unit	Formation	Description
Upper Three Runs Aquifer	"Upland Unit"	Poorly sorted, clayey-to-silty sands, with lenses and layers of conglomerates, pebbly sands, and clays. Clay clasts are abundant, and cross-bedding and flecks of weathered feldspar are locally common.
	Tobacco Road Formation	Moderately to poorly sorted, variably colored, fine-to-coarse grained sand, pebbly sand, and minor clay beds
	Dry Branch Formation	Variably colored, poorly sorted to well sorted sand with interbedded tan to gray clay
	Clinchfield Formation	Light colored basal quartz sand and glauconitic, biomoldic limestone, calcareous sand and clay. Sand beds of the formation constitute Riggins Mill Member and consist of medium to coarse, poorly to well sorted, loose and slightly indurated, tan, gray, and green quartz. The carbonate sequence of the Clinchfield consists of Utley Member -- sandy, glauconitic limestone and calcareous sand with indurated biomoldic facies
	Tinker/Santee Formation	Unconsolidated, moderately sorted, subangular, lower coarse-to-medium grained, slightly gravely, immature yellow and tan quartz sand and clayey sand; calcareous sands and clays and limestone also occur in F- and H-Areas.
Gordon Confining Unit (green clay)	Blue Bluff Member of Santee Limestone	Micritic limestone
	Warley Hill Formation	Fine grained, glauconitic, clayey sand, and clay that thicken, thin, and pinch out abruptly
Gordon Aquifer	Congaree Formation	Yellow, orange, tan, gray, and greenish gray, well-sorted, fine-to-coarse-grained quartz sands. Thin clay laminae occur throughout the section, with pebbly layers, clay clasts, and glauconite in places. In some places on SRS, upper part of Congaree Formation is cemented with silica; in other places it is slightly calcareous. Glauconitic clay, encountered in some borings on SRS near the base of this formation, indicates that basal contact is unconformable
	Fourmile Formation	Tan, yellow-orange, brown, and white, moderately to well-sorted sand, with clay beds near middle and top of unit. The sand is very coarse to fine-grained, with pebbly zones common. Glauconite and dino-flagellate fossils occur.
	Snapp Formation	Silty, medium- to coarse-grained quartz sand interbedded with clay. Dark, micaceous, lignitic sand also occurs. In northwestern part of SRS, this Formation is less silty and better sorted, with thinner clay interbeds.

a. Source: Aadland, Gellici, and Thayer (1995).

3.1.2 SUBSURFACE FEATURES

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There are several fault systems off the Site northwest of the Fall Line (DOE 1990a). A more recent study of geophysical evidence (Wike, Moore-Shedrow, and Shedrow 1996) and an earlier study (Stephenson and Stieve 1992) identified the faults indicated on Figure 3.1-3. The earlier study identified the following faults – Pen Branch, Steel Creek, Advanced Tactical Training Area (ATTA), Crackerneck, Ellenton, and Upper Three Runs – under SRS. The one closest to the areas under consideration is the Steel Creek Fault, which passes through L Area and is approximately 1 mile (1.6 kilometers) northwest of P Area. The Upper Three Runs Fault, which is a Paleozoic fault that does not cut Coastal Plain sediments, passes approximately 1 mile (1.6 kilometers) from F Area. The lines shown on Figure 3.1-3 represent the projection of faults to the ground surface. The actual faults do not reach the surface, but rather stop several hundred feet below.

Based on the available information, none of the faults discussed in this section is capable, which means that it has not moved at or near the ground surface within the past 35,000 years or is associated with another fault that had moved in the past 35,000 years. (10 CFR 100 contains a more detailed definition of a capable fault.)

3.1.3 SEISMICITY

Two major earthquakes have occurred within 186 miles (300 kilometers) of SRS.

- The Charleston, South Carolina, earthquake of 1886 had an estimated Richter scale magnitude of 6.8; it occurred approximately 90 miles (145 kilometers) from the SRS area, which experienced an estimated peak horizontal acceleration of 10 percent of gravity (0.10g) (URS/Blume 1982).
- The Union County, South Carolina, earthquake of 1913 had an estimated Richter scale magnitude of 6.0 and occurred about 99 miles (160 kilometers) from the Site (Bollinger 1973).

Because these earthquakes are not associated conclusively with a specific fault, researchers cannot determine the amount of displacement resulting from them.

In recent years, three earthquakes occurred inside the SRS boundary as reported by local print and media and cited in DOE (1999a).

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- On May 17, 1997, with a Richter scale magnitude of 2.3 and a focal depth of 3.38 miles (5.44 kilometers); its epicenter was south-east of K Area.
- On August 5, 1988, with a local Richter scale magnitude of 2.0 and a focal depth of 1.66 miles (2.68 kilometers); its epicenter was northeast of K Area.
- On June 8, 1985, with a local Richter scale magnitude of 2.6 and a focal depth of 0.59 mile (0.96 kilometer); its epicenter was south of C Area and west of K Area.

Existing information does not relate these earthquakes conclusively with known faults under the Site. Figure 3.1-3 shows the locations of the epicenters of these earthquakes.

Outside the SRS boundary, an earthquake with a Richter scale magnitude of 3.2 occurred on August 8, 1993, approximately 10 miles (16 kilometers) east of the City of Aiken near Coughton, South Carolina. People reported feeling this earthquake in Aiken, New Ellenton (immediately north of SRS), North Augusta [approximately 25 miles (40 kilometers) northwest of the SRS], and on the Site (Aiken Standard 1993).

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3.2 Water Resources

3.2.1 SURFACE WATER RESOURCES

This section describes the surface water, and the quality of that water, in the area potentially affected by the proposed action, including the Savannah River, Upper Three Runs, Fourmile Branch, and Steel Creek.

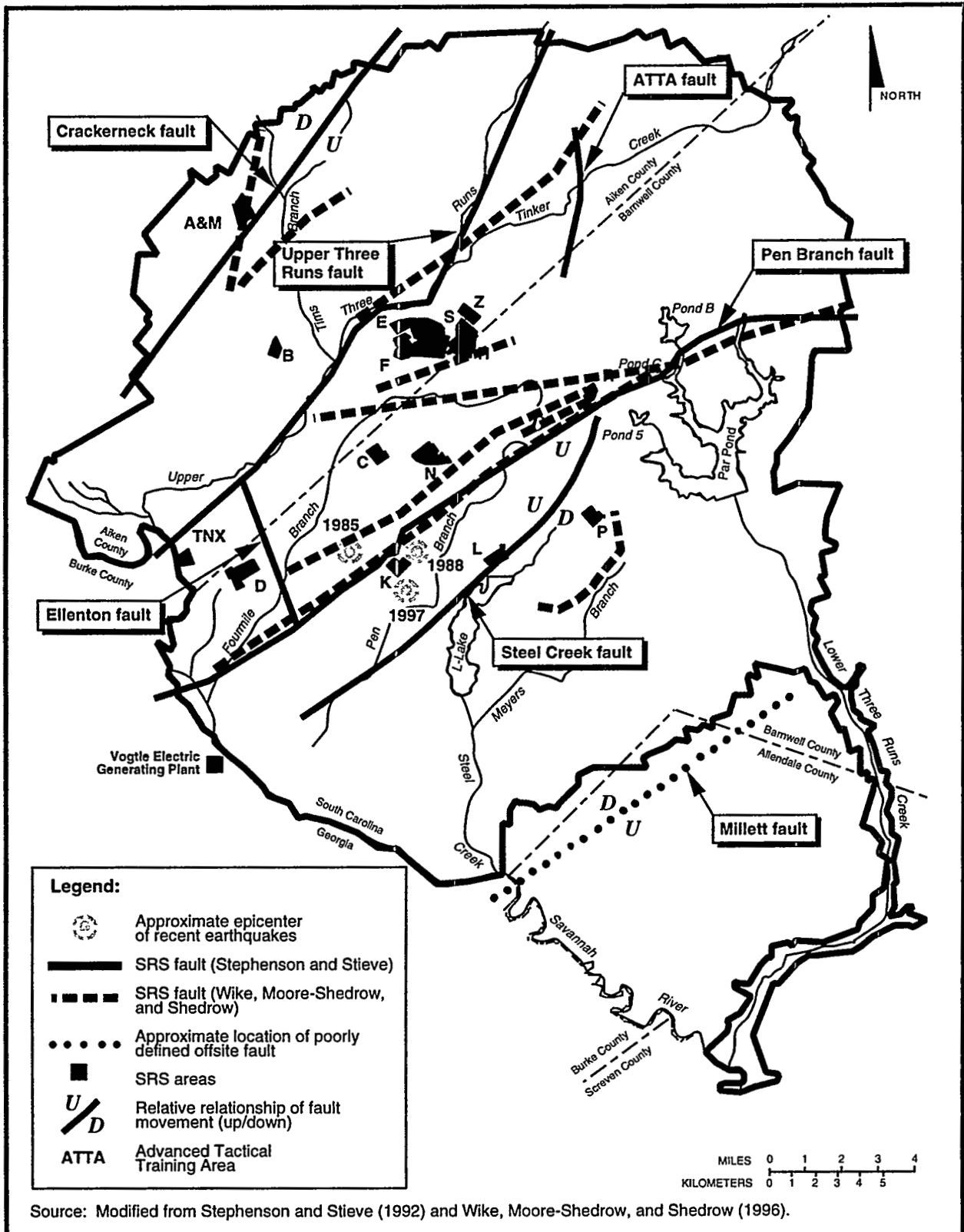


Figure 3.1-3. Savannah River Site, showing seismic fault lines and locations of onsite earthquakes and their year of occurrence.

3.2.1.1 Savannah River

The Savannah River bounds SRS on its southwestern border for about 20 miles (32 kilometers), approximately 160 river miles (260 river kilometers) from the Atlantic Ocean. Five upstream reservoirs -- Jocassee, Keowee, Hartwell, Richard B. Russell, and Strom Thurmond -- minimize the effects of droughts and the impacts of low flow on downstream water quality and fish and wildlife resources in the river. River flow averages about 10,000 cubic feet (283 cubic meters) per second at SRS (DOE 1995a).

The Savannah River, which forms the boundary between Georgia and South Carolina, supplies potable water to a number of users. Upstream of SRS, the river supplies domestic and industrial water for Augusta, Georgia, and North Augusta, South Carolina. Approximately 130 river miles (210 river kilometers) downstream of SRS, the river supplies domestic and industrial water for Savannah, Georgia, and Beaufort and Jasper Counties in South Carolina through intakes at about River Mile 29 and River Mile 39, respectively (DOE 1995b).

The Savannah River receives sewage treatment plant effluent from Augusta, Georgia; North Augusta, Aiken, and Horse Creek Valley, South Carolina; and from a number of SRS operations through discharges to onsite streams. In addition, the Georgia Power Company's Vogtle Electric Generating Plant withdraws an average of 46 cubic feet (1.3 cubic meters) per second for cooling and returns an average of 12 cubic feet (0.35 cubic meter) per second of cooling tower blowdown. The Urquhart Steam Generating Station at Beech Island, South Carolina, withdraws approximately 265 cubic feet (7.5 cubic meters) per second for once-through cooling water (DOE 1995a).

On SRS, a swamp occupies the floodplain along the Savannah River for approximately 10 miles (17 kilometers); the swamp is about 1.5 miles (2.5 kilometers) wide. A natural levee separates the river from the floodplain. Figure 3.2-1 shows the 100-year floodplain of the Savannah

River in the SRS vicinity and the floodplains of major tributaries that drain the Site (DOE 1995a).

3.2.1.2 SRS Streams

Five tributaries of the Savannah River -- Upper Three Runs, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs -- drain almost all of the SRS (Figure 3.2-1). Each stream originates on the Aiken Plateau in the Coastal Plain and descends 50 to 200 feet (15 to 60 meters) before discharging into the river. The streams, which historically received varying amounts of effluent from SRS operations, are not commercial sources of water. Their natural flows range from less than 10 cubic feet (1 cubic meter) per second in smaller streams such as Pen Branch to 240 cubic feet (6.8 cubic meters) per second in Upper Three Runs (DOE 1995a).

Upper Three Runs, Fourmile Branch, and Steel Creek are the streams closest to most SRS spent nuclear fuel management locations (see Figure 3.2-1). These streams also are closest to the areas where DOE is most likely to place new spent nuclear fuel facilities.

Upper Three Runs is a large, cool, blackwater stream in the northern part of SRS. It drains an area of approximately 210 square miles (545 square kilometers), and has an average discharge of 330 cubic feet (9.3 cubic meters) per second at its mouth. Upper Three Runs is approximately 25 miles (40 kilometers) long, with its lower 17 miles (28 kilometers) inside SRS boundaries. This creek receives more water from underground sources than other SRS streams and, therefore, has lower conductivity, hardness, and pH values. Upper Three Runs is the only major tributary on SRS that has never received thermal discharges from nuclear reactors (DOE 1995a).

Fourmile Branch is about 15 miles (24 kilometers) long and drains an area of approximately 22 square miles (57 square kilometers). At its headwaters, Fourmile Branch is a small blackwater stream that currently receives

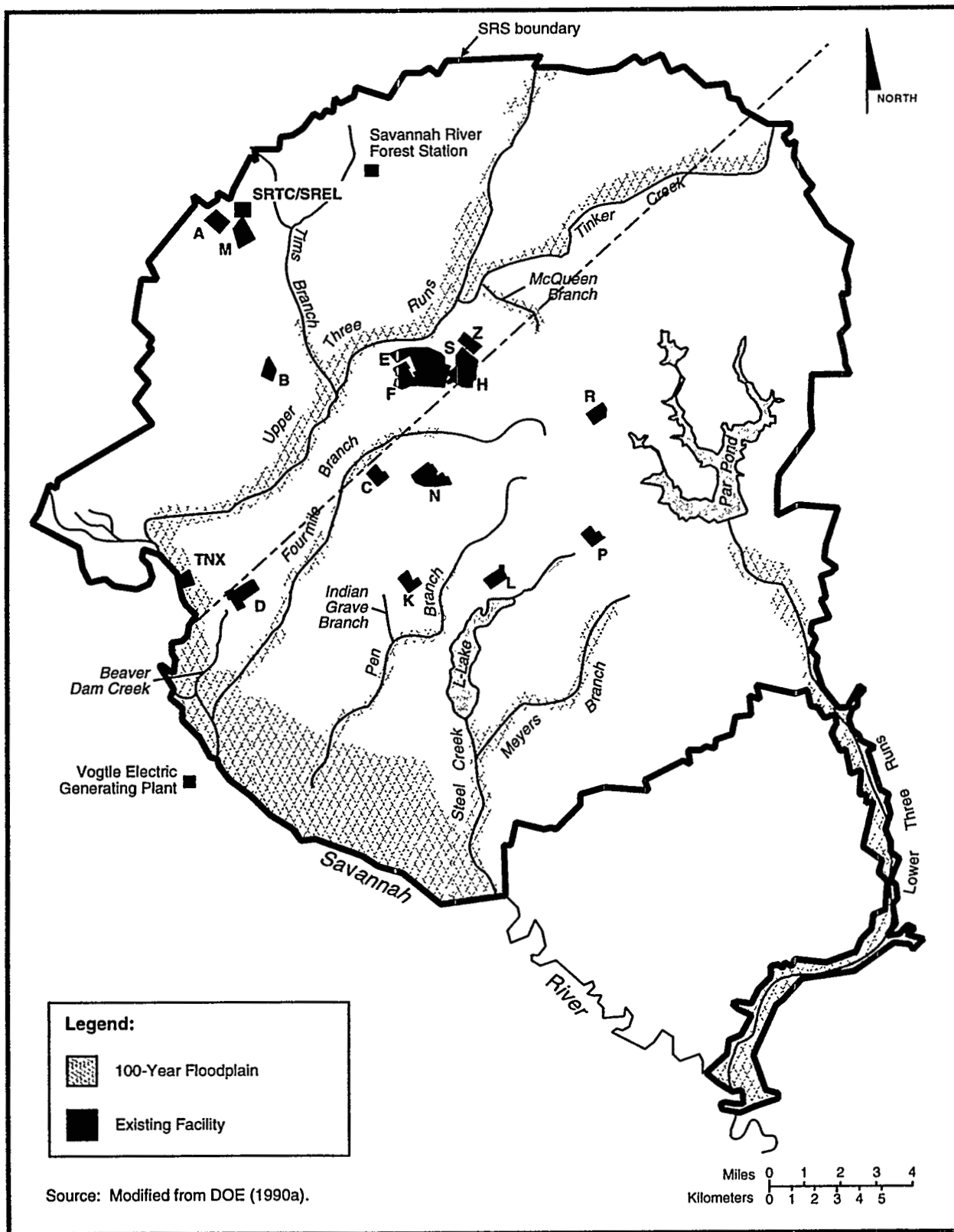


Figure 3.2-1. Savannah River Site, showing 100-year floodplain and major stream systems.

impacts from SRS operations. The water chemistry in the headwater area is very similar to that of Upper Three Runs, with the exception of nitrate concentrations, which are an order of magnitude higher than those in Upper Three Runs (DOE 1995a). These elevated concentrations are probably the result of groundwater transport and outcropping from the F- and H-Area seepage basins. In its lower reaches, Fourmile Branch broadens and flows through a delta formed by the deposition of sediments. Although most of the flow through the delta is in one main channel, the delta has many standing dead trees, logs, stumps, and cypress trees that provide structure and reduce the water velocity in some areas. Downstream of the delta, the creek flows in one main channel and discharges primarily into the Savannah River at River Mile 152, while a small portion flows west and enters Beaver Dam Creek, a small onsite tributary of the Savannah River (DOE 1995a).

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Steel Creek is about 9 miles (15 kilometers) long and, with Meyers Branch, drains an area of approximately 35 square miles (90 square kilometers) (DOE 1996a). Its headwaters originate near P Reactor. The creek flows southwest about 2 miles (3 kilometers) before it enters the headwaters of L Lake. Flow from the outfall of the L-Lake dam travels about 3 miles (5 kilometers) before entering the Savannah River swamp and then another 2 miles (3 kilometers) before entering the river.

Meyers Branch, the main tributary of Steel Creek, flows approximately 6 miles (10 kilometers) before entering Steel Creek. Meyers Branch is a small blackwater stream that has remained relatively undisturbed by SRS operations. The confluence of Meyers Branch and Steel Creek is downstream from the L-Lake dam. Steel Creek received intermittent thermal effluent from P and L Reactors from 1954 to 1964, and from L Reactor only from 1964 to 1968 (Halverson et al. 1997). Effluents from L and P Areas flow to L Lake and subsequently to Steel Creek through the L-Lake dam outfall. During water year 1996, flows in Steel Creek (downstream of the confluence with Meyers

Branch) averaged 59.2 cubic feet (1.7 cubic meters) per second (DOE 1996a).

3.2.1.3 Surface-Water Quality

In 1996, releases of radionuclides from the SRS to surface waters amounted to 8,550 curies of tritium, 0.214 curie of strontium-89 and -90, and 0.05 curie of plutonium-239 (Arnett and Mamatey 1998a). Table 3.2-1 lists radioactive liquid releases by source for 1997; Table 3.2-2 lists radioactive liquid releases by outfall or facility and compares annual average radionuclide concentrations to DOE concentration guides (Figure 3.2-2 shows outfall and facility locations for radioactive surveillance). The resulting doses to a downriver consumer of river water from radionuclides released from the Site were less than 2 percent of the U.S. Environmental Protection Agency (EPA) and DOE standards for public water supplies (40 CFR Part 141 and DOE Order 5400.5, respectively) and less than 0.2 percent of the DOE dose standard from all pathways (DOE 1990b; Arnett and Mamatey 1998).

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The South Carolina Department of Health and Environmental Control (SCDHEC) regulates the physical properties and concentrations of chemicals and metals in SRS effluents under the National Pollutant Discharge Elimination System (NPDES) program. SCDHEC, which also regulates biological water quality standards for SRS waters, has classified the Savannah River and SRS streams as "Freshwaters." In 1997, 99.9 percent of the NPDES water quality analyses on SRS effluents were in compliance with the SRS NPDES permit; only 7 of 5,758 analyses exceeded permit limits (Arnett and Mamatey 1998a). A comparison of 1997 Savannah River water quality analysis upstream and downstream of SRS showed no significant differences, and a comparison with historical data indicates that coliform data are within normal fluctuation for river water in this area and the overall exceedances decreased in number from 1996 (Arnett and Mamatey 1998a). Table 3.2-3 summarizes the water quality of Fourmile Creek, Steel Creek, and Upper Three Runs for 1996.

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Table 3.2-1. Annual liquid releases by source for 1997 (including direct and seepage basin migration releases).^a

Radionuclide ^b	Half-life (years)	Curies					
		Reactors	Separations ^c	Reactor materials	TNX	SRTC	Total
H-3 (oxide)	12.3	2.91×10 ³	5.24×10 ⁻³		4.02×10 ²	1.82	8.55×10 ⁻³
Sr-89,90 ^d	29.1	6.46×10 ⁻²	1.40×10 ⁻¹		5.09×10 ⁻³	4.10×10 ⁻³	2.14×10 ⁻¹
I-129 ^e	1.6×10 ⁷		7.82×10 ^{-2e}				7.82×10 ^{-2d}
Cs-137	30.2	2.86×10 ⁻³	4.49×10 ⁻²				4.78×10 ⁻²
U-234	2.46×10 ⁵	4.45×10 ⁻³	2.30×10 ⁻²	2.68×10 ⁻⁵	1.52×10 ⁻⁶	1.06×10 ⁻⁴	2.76×10 ⁻²
U-235	7.04×10 ⁸	4.91×10 ⁻⁵	7.23×10 ⁻⁴		1.37×10 ⁻⁷	3.44×10 ⁻⁶	7.76×10 ⁻⁴
U-238	4.47×10 ⁹	3.83×10 ⁻³	2.57×10 ⁻²	5.71×10 ⁻⁵	9.19×10 ⁻⁶	1.11×10 ⁻⁴	2.97×10 ⁻²
Pu-238	87.7	4.24×10 ⁻⁵	9.57×10 ⁻⁴		7.68×10 ⁻⁷	1.78×10 ⁻⁶	1.00×10 ⁻³
Pu-239 ^f	24,100	1.10×10 ⁻²	3.39×10 ⁻²	1.14×10 ⁻³	1.12×10 ⁻³	3.38×10 ⁻³	5.05×10 ⁻²
Am-241	432.7		7.81×10 ⁻⁶	2.11×10 ⁻⁶			9.92×10 ⁻⁶
Cm-244	18.1		2.93×10 ⁻⁶	4.14×10 ⁻⁷			3.34×10 ⁻⁶

Notes: Blank spaces indicate no quantifiable activity.

a. Source: Arnett and Mamatey (1998a).

b. H = hydrogen (H-3 = tritium), Sr = strontium, I = iodine, Cs = cesium, U = uranium, Pu = plutonium, Am = americium, Cm = curium.

c. Includes separations, waste management, and tritium facilities.

d. Includes unidentified beta.

e. Measured I-129 doses were not available for 1997. The value for separations emissions is from 1996.

f. Includes unidentified alpha.

TNX = a technology development facility adjacent to the Savannah River.

SRTC = Savannah River Technology Center.

(Figure 3.2-3 shows stream water quality monitoring locations.)

Certain technologies, including those considered in this EIS, generate liquid byproducts that are transferred to the F- and H-Area Tank Farms. Evaporator overheads from these tanks are condensed and treated at the F- and H-Area Effluent Treatment Facility (ETF). Waste concentrate from the ETF is disposed of in the Z-Area Saltstone Manufacturing and Disposal Facility and the decontaminated wastewater is discharged to Upper Three Runs through NPDES outfall H-16. These existing facilities are described in the *Interim Management of Nuclear Materials EIS* (DOE 1995b) and the *Defense Waste Processing Facility Supplemental EIS* (DOE 1994). Requirements for spent nuclear fuel processing are included in these documents and, therefore, this EIS considers those facilities and processed waste amounts to be part of the SRS baseline.

3.2.2 GROUNDWATER RESOURCES

3.2.2.1 Groundwater Features

In the SRS region, the subsurface contains two hydrogeologic provinces. The uppermost, consisting of a wedge of unconsolidated Coastal Plain sediments of Late Cretaceous and Tertiary age, is the Atlantic Coastal Plain Hydrogeologic Province. Beneath the sediments of the Atlantic Coastal Plain Hydrogeologic Province are rocks of the Piedmont Hydrogeologic Province. These rocks consist of Paleozoic igneous and metamorphic basement rocks and lithified mudstone, sandstone, and conglomerates of the Dunbarton basin of the Upper Triassic. Sediments of the Atlantic Coastal Plain Hydrogeologic Province are divided into three main

Table 3.2-2. Liquid radioactive releases by outfall/facility and comparison of annual average radionuclide concentrations to DOE derived concentration guides.^a

Outfall or Facility	Radionuclide ^b	Quantity of Radionuclides Released during 1997 (Ci)	Average Effluent Concentration during 1997 (μCi/mL)	DOE DCGs ^c (μCi/mL)
C Area (C Reactor)				
C Canal	H-3 (oxide)	1.20	1.75×10 ⁻⁶	2.00×10 ⁻³
	Sr-89,90	Below MDL	ND	1.00×10 ⁻⁶
	Cs-137	Below MDL	1.02×10 ⁻⁹	3.00×10 ⁻⁶
F Area (Separations and Waste Management)				
F-01	H-3 (oxide)	5.03×10 ⁻²	2.54×10 ⁻⁷	2.00×10 ⁻³
	Sr-89,90	Below MDL	ND	1.00×10 ⁻⁶
	Cs-137	Below MDL	1.32×10 ⁻⁹	3.00×10 ⁻⁶
F-012 (281-8F Retention Basin)	H-3 (oxide)	7.67×10 ⁻¹	9.83×10 ⁻⁶	2.00×10 ⁻³
	Sr-89,90	Below MDL	3.01×10 ⁻⁹	1.00×10 ⁻⁶
	Cs-137	158×10 ⁻³	2.07×10 ⁻⁸	3.00×10 ⁻⁶
	H-3 (oxide)	1.73×10 ⁻²	1.63×10 ⁻⁶	2.00×10 ⁻³
	Sr-89,90	3.13×10 ⁻⁵	4.39×10 ⁻⁹	1.00×10 ⁻⁶
	Cs-137	5.92×10 ⁻⁴	2.30×10 ⁻⁸	3.00×10 ⁻⁶
Fourmile Branch-3 (F-Area Effluent)	H-3 (oxide)	1.32×10	7.80×10 ⁻⁷	2.00×10 ⁻³
	Sr-89,90	Below MDL	4.16×10 ⁻¹⁰	1.00×10 ⁻⁶
	Cs-137	Below MDL	8.97×10 ⁻¹⁰	3.00×10 ⁻⁶
Upper Three Runs-2 (F Storm Sewer)	H-3 (oxide)	1.66×10 ⁻¹	8.78×10 ⁻⁷	2.00×10 ⁻³
	Sr-89,90	Below MDL	8.56×10 ⁻¹¹	1.00×10 ⁻⁶
	Cs-137	Below MDL	5.13×10 ⁻¹⁰	3.00×10 ⁻⁶
	U-234	6.86×10 ⁻⁵	3.48×10 ⁻¹⁰	6.00×10 ⁻⁷
	U-235	5.15×10 ⁻⁶	3.02×10 ⁻¹¹	6.00×10 ⁻⁷
	U-238	1.90×10 ⁻⁴	9.15×10 ⁻¹⁰	6.00×10 ⁻⁷
	Pu-238	1.54×10 ⁻⁵	9.10×10 ⁻¹¹	4.00×10 ⁻⁸
	Pu-239	7.73×10 ⁻⁶	4.66×10 ⁻¹¹	3.00×10 ⁻⁸
	Am-241	7.77×10 ⁻⁶	3.98×10 ⁻¹¹	3.00×10 ⁻⁸
	Cm-244	2.92×10 ⁻⁶	1.74×10 ⁻¹¹	6.00×10 ⁻⁸
Upper Three Runs F-3 (Naval Fuel Effluent)	H-3 (oxide)	3.45×10 ⁻²	1.46×10 ⁻⁶	2.00×10 ⁻³
	Sr-89,90	Below MDL	1.16×10 ⁻¹⁰	1.00×10 ⁻⁶
	Cs-137	Below MDL	2.47×10 ⁻¹⁰	3.00×10 ⁻⁶
	U-234	1.62×10 ⁻⁵	8.95×10 ⁻¹⁰	6.00×10 ⁻⁷
	U-235	5.86×10 ⁻⁶	2.30×10 ⁻⁹	6.00×10 ⁻⁷
	U-238	3.04×10 ⁻⁶	1.76×10 ⁻¹⁰	6.00×10 ⁻⁷
	Pu-238	1.61×10 ⁻⁷	6.23×10 ⁻¹²	4.00×10 ⁻⁸
	Pu-239	2.60×10 ⁻⁸	5.04×10 ⁻¹²	3.00×10 ⁻⁸
	Am-241	4.49×10 ⁻⁸	7.07×10 ⁻¹³	3.00×10 ⁻⁸
	Cm-244	9.54×10 ⁻⁹	6.84×10 ⁻¹¹	6.00×10 ⁻⁸

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Table 3.2-2. (continued).

Outfall or Facility	Radionuclide ^b	Quantity of Radionuclides Released during 1997 (Ci)	Average Effluent Concentration during 1997 (μCi/mL)	DOE DCGs ^c (μCi/mL)
H Area (Separations and Waste Management)				
Fourmile Branch-1C (H-Area Effluent)	H-3 (oxide)	3.85×10	9.22×10 ⁻⁶	2.00×10 ⁻³
	Sr-89,90	7.93×10 ⁻⁵	7.05×10 ⁻¹⁰	1.00×10 ⁻⁶
	Cs-137	6.77×10 ⁻⁴	3.27×10 ⁻⁹	3.00×10 ⁻⁶
	H-3 (oxide)	4.96×10 ⁻¹	1.23×10 ⁻⁵	2.00×10 ⁻³
	Sr-89,90	3.48×10 ⁻⁶	5.40×10 ⁻¹⁰	1.00×10 ⁻⁶
	Cs-137	2.15×10 ⁻⁶	7.15×10 ⁻¹⁰	3.00×10 ⁻⁶
	U-234	2.77×10 ⁻⁶	8.54×10 ⁻¹¹	6.00×10 ⁻⁷
	U-235	9.84×10 ⁻⁹	8.61×10 ⁻¹²	6.00×10 ⁻⁷
	U-238	2.07×10 ⁻⁶	6.58×10 ⁻¹¹	6.00×10 ⁻⁷
	Pu-238	5.09×10 ⁻⁷	2.45×10 ⁻¹¹	4.00×10 ⁻⁸
H-017 (281-8H Retention Basin)	Pu-239	8.93×10 ⁻⁸	6.37×10 ⁻¹²	3.00×10 ⁻⁸
	H-3	7.17×10 ⁻¹	1.02×10 ⁻⁵	2.00×10 ⁻³
	Sr-89,90	5.21×10 ⁻⁴	7.91×10 ⁻⁹	1.00×10 ⁻⁶
H-018 (200-H Cooling Basin)	Cs-137	1.04×10 ⁻²	1.11×10 ⁻⁷	3.00×10 ⁻⁶
	H-3 (oxide)	1.44×10 ⁻¹	2.27×10 ⁻⁵	2.00×10 ⁻³
	Sr-89,90	2.75×10 ⁻⁴	4.58×10 ⁻⁸	1.00×10 ⁻⁶
HP-15 (Tritium Facility Outfall)	Cs-137	2.21×10 ⁻³	3.71×10 ⁻⁷	3.00×10 ⁻⁶
	H-3 (oxide)	1.74×10	1.55×10 ⁻⁵	2.00×10 ⁻³
	Cs-137	Below MDL	7.75×10 ⁻¹¹	3.00×10 ⁻⁶
HP-52 (H-Area Tank Farm)	H-3 (oxide)	2.43×10	1.30×10 ⁻⁶	2.00×10 ⁻³
	SR-89,90	Below MDL	7.67×10 ⁻¹¹	1.00×10 ⁻⁶
	Cs-137	1.58×10 ⁻⁴	1.92×10 ⁻⁹	3.00×10 ⁻⁶
McQueen's Branch at Rd F	H-3 (oxide)	120×10 ¹	1.05×10 ⁻⁵	2.00×10 ⁻³
	Cs-137	Below MDL	4.85×10 ⁻¹⁰	3.00×10 ⁻⁶
	Upper Three Runs – 2A (ETF ^e Outfall at Rd C)	H-3 (oxide)	(f)	2.00×10 ⁻³
L Area (L Reactor)	Sr-89,90	1.28×10 ⁻⁵	2.24×10 ⁻⁹	1.00×10 ⁻⁶
	Cs-137	1.79×10 ⁻²	2.16×10 ⁻⁷	3.00×10 ⁻⁶
	Cs-137	6.02×10	3.38×10 ⁻⁷	2.00×10 ⁻³
L-007	Sr-89,90	Below MDL	1.16×10 ⁻¹⁰	1.00×10 ⁻⁶
	Cs-137	Below MDL	4.53×10 ⁻¹⁰	3.00×10 ⁻⁶
P Area (P Reactor)				
P-013A	H-3 (oxide)	7.18×10 ⁻¹	2.96×10 ⁻⁴	2.00×10 ⁻³
	Sr-89,90	5.25×10 ⁻⁶	3.47×10 ⁻⁹	1.00×10 ⁻⁶
	Cs-137	2.38×10 ⁻⁴	9.86×10 ⁻⁸	3.00×10 ⁻⁶
P-019A (P-Area Canal Par Pond)	H-3 (oxide)	3.25×10 ⁻¹	5.41×10 ⁻⁷	2.00×10 ⁻³
	Sr-89,90	Below MDL	3.03×10 ⁻¹⁰	1.00×10 ⁻⁶
	Cs-137	Below MDL	ND	3.00×10 ⁻⁶

a. Source: Arnett and Mamatey (1998a).

b. H = hydrogen (H-3 = tritium), Sr = strontium, I = iodine, Cs = cesium, U = uranium, Pu = plutonium, Am = americium, Cm = curium.

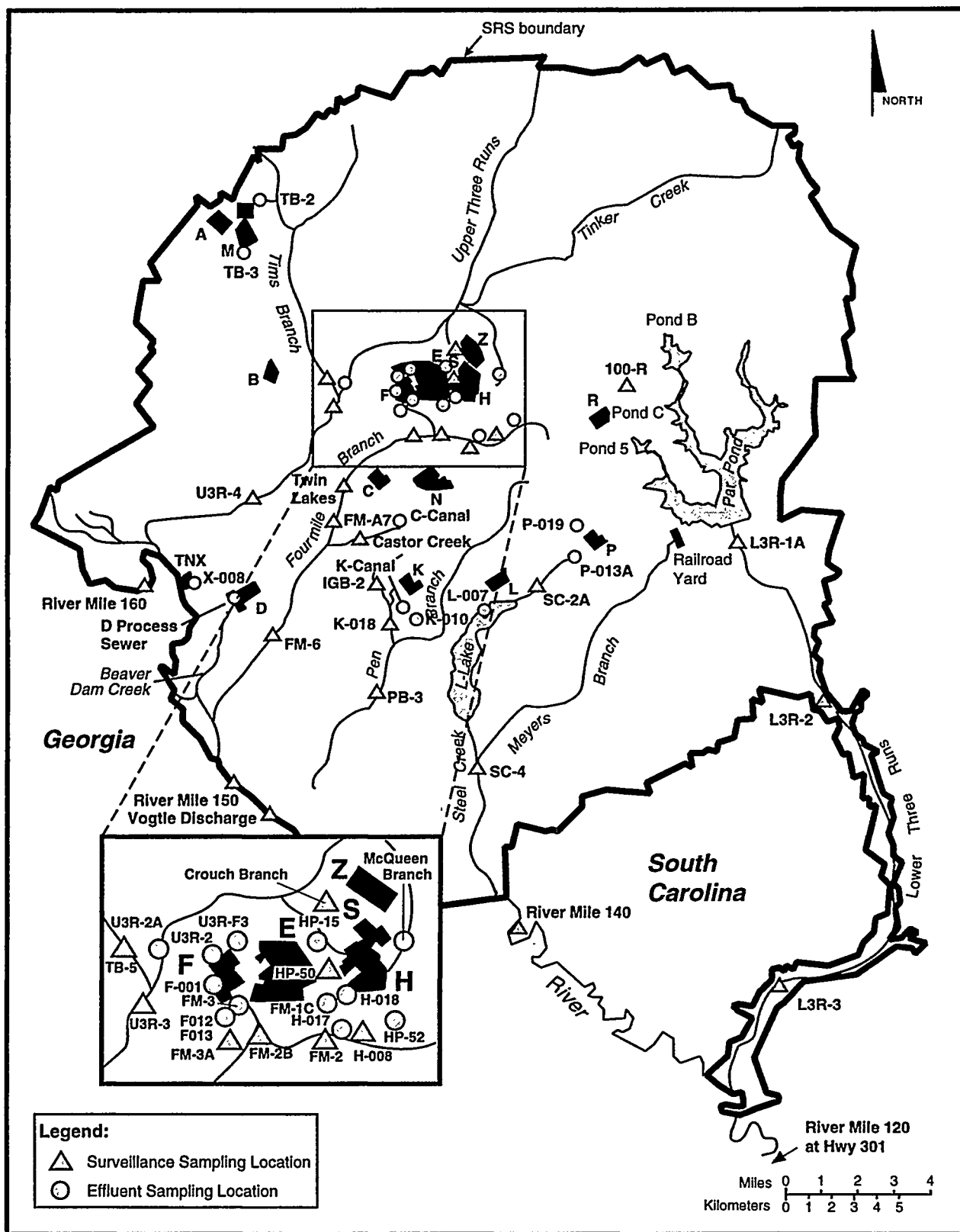
c. DCG = derived concentration guide. Source: DOE Order 5400.5. In cases where different chemical forms have different DCGs, the lowest DCG for the radionuclide is given. DCGs are defined as the concentration of that radionuclide that will give a 50-year committed effective dose equivalent of 100 mrem under conditions of continuous exposure for one year. DCGs are reference values only and are not considered release limits or standards.

d. MDL = minimum detectable level.

e. ETF = Effluent Treatment Facility.

f. Outfall concentrations for tritium exceed the DCG guidelines. DOE Order 5400.5 exempts tritium from "best available technology" requirements because there is no practical technology available for removing tritium from dilute liquid waste streams.

ND = not detected.



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Figure 3.2-2. Radiological surface-water sampling locations.

Table 3.2-3. SRS stream water quality (onsite downstream locations).^a

Parameter ^b	Units	Fourmile Branch (FM-6) average	Steel Creek (SC-4) average	Upper Three Runs (U3R-4) average	Water Quality Criterion ^c , MCL ^d , or DCG ^e
Aluminum	Mg/L	0.200 ^f	0.018	0.274 ^f	0.087
Cadmium	Mg/L	ND ^g	ND	ND	0.00066
Calcium	Mg/L	2.94	2.53	1.62	NA ^h
Cesium-137	pCi/L	NR ⁱ	NR	NR	120 ^e
Chromium	mg/L	ND	ND	ND	0.011
Copper	mg/L	0.015 ^f	0.028 ^f	0.036 ^f	0.0065
Dissolved oxygen	mg/L	7.9	8.73	8.2	≥5
Iron	mg/L	0.69	0.349	0.586	1
Lead	mg/L	ND	ND	ND	0.0013
Magnesium	mg/L	0.659 ^f	0.854 ^f	0.385 ^f	0.3
Manganese	mg/L	0.055	0.048	0.026	1
Mercury	mg/L	ND	0.0002	ND	0.000012
Nickel	mg/L	0.01	0.01	0.012	0.088
Nitrate (as nitrogen)	mg/L	1.36	0.16	0.24	10 ^d
pH	pH	6.31	6.32	6.3	6-8.5
Plutonium-238	pCi/L	NR	NR	NR	1.6 ^e
Plutonium-239	pCi/L	NR	NR	NR	1.2 ^e
Sodium	mg/L	6.8	1.89	1.58	NA
Strontium-89,90	pCi/L	NR	NR	NR	8 ^d
Suspended solids	mg/L	8.08	5.2	14.1	NA
Temperature ^j	°C	18.1	18.6	17.3	32.2
Total dissolved solids	mg/L	355.6	48	36	500 ^k
Tritium	pCi/L	NR	NR	NR	20,000 ^d
Uranium-234	pCi/L	NR	NR	NR	20 ^e
Uranium-235	pCi/L	NR	NR	NR	24 ^e
Uranium-238	pCi/L	NR	NR	NR	24 ^e
Zinc	mg/L	0.041	0.040	0.028	0.059

a. Source: Arnett and Mamatey (1997).

b. Parameters DOE routinely measures as a regulatory requirement or as part of ongoing monitoring programs.

c. Water Quality Criterion (WQC) is Aquatic Chronic Toxicity unless otherwise indicated.

d. MCL = Maximum Contaminant Level; State Primary Drinking Water Regulations.

e. DCG = DOE Derived Concentration Guides for Water (DOE Order 5400.5). DCG values are based on committed effective dose of 100 millirem per year; however, because drinking water MCL is based on 4 millirem per year, value listed is 4 percent of DCG.

f. Concentration exceeded WQC; however, these criteria are for comparison only. WQCs are not legally enforceable.

g. ND = Not Detected.

h. NA = Not Applicable.

i. NR = Not Reported.

j. Shall not be increased more than 2.8°C (5°F) above natural temperature conditions or exceed a maximum of 32.2°C (90°F) as a result of the discharge of heated liquids unless appropriate temperature criterion mixing zone has been established.

k. Secondary MCL; State Primary Drinking Water Regulations.



Figure 3.2-3. SRS streams and Savannah River water quality sampling locations.

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aquifer systems, the Floridan Aquifer System, the Dublin Aquifer System, and the Midville Aquifer System as shown in Figure 3.1-2 (Aadland, Gellici, and Thayer 1995). Each aquifer system is divided from the others by two confining systems, the Meyers Branch Confining System and the Allendale Confining System, as shown in Figure 3.1-2.

Groundwater within the Floridan system (the shallow aquifer beneath the Site) flows slowly toward SRS streams and swamps and into the Savannah River at rates ranging from inches to several hundred feet per year. The depth to which onsite streams cut into soils and the orientation of the soil formations control the horizontal and vertical movement of the groundwater. The valleys of smaller perennial streams allow discharge from the shallow saturated geologic formations. The valleys of major tributaries of the Savannah River (e.g., Upper Three Runs) drain formations of intermediate depth, and the river valley drains deep formations. With the release of water to the streams, the hydraulic head of the aquifer unit releasing the water can become less than that of the underlying unit. If this occurs, groundwater has the potential to migrate from the lower unit to the overlying unit.

Groundwater flow in the shallow aquifer (Floridan) system is vertically downward in the divide areas between surface water drainages due to the decreasing hydraulic head with increasing depth. In areas along the lower reaches of most of the Site streams, groundwater moves vertically upward from deeper aquifers to the shallow aquifers. In these areas hydraulic heads increase with depth.

In the vicinity of these streams, the vertical upward flow occurs across the Crouch Branch Confining Unit/Gordon Confining Unit. At these locations any contaminants in the overlying aquifer system are prevented from migrating into deeper aquifers by the prevailing hydraulic gradient and the low permeability of the confining unit. Horizontal groundwater flow occurs at the M-Area metallurgical laboratory (to the west-northwest in the shallow aquifer and sub-

sequent flow to the south toward Upper Three Runs in the intermediate aquifer), K-Area Disassembly Basin (toward Pen Branch and L Lake), P-Area Disassembly Basin (toward Steel Creek), F Canyon (toward Upper Three Runs and Fourmile Branch), and H Canyon (toward Upper Three Runs and its tributaries).

3.2.2.2 Groundwater Use

Groundwater is a domestic, municipal, and industrial water source throughout the Upper Coastal Plain. Domestic water supplies come primarily from the shallow aquifers including the Gordon Aquifer and the Upper Three Runs Aquifer (water-table aquifer). Most municipal and industrial water supplies in Aiken County are from the Cretaceous intermediate to deep aquifer units. In Barnwell and Allendale Counties some municipal water supplies are from the Gordon Aquifer and overlying units that thicken to the southeast. At SRS, most groundwater production for domestic and process water comes from the intermediate/deep aquifers (i.e., the Crouch Branch and McQueen Branch Aquifers), with a few lower-capacity process water wells pumping from the shallower Gordon Aquifer.

Every major operating area at SRS has groundwater wells; total groundwater production ranges from 9 to 12 million gallons (34,000 to 45,000 cubic meters) per day, similar to the volume pumped for industrial and municipal production within 10 miles (16 kilometers) of the Site (Arnett and Mamatey 1996).

From October 1995 to September 1996, the total groundwater withdrawal rate for C, F, H, P, and L Areas was approximately 4 million gallons (15,130 cubic meters) per day. Groundwater in C Area comes from two domestic wells that produced approximately 220,000 gallons (830 cubic meters) per day. Groundwater in F Area is pumped from four process production and two domestic wells. The total F-Area groundwater production rate from October 1995 to September 1996 was approximately 1.58 million gallons (5,981 cubic meters) per day. During the same period, wells in H, L, and P Areas

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produced approximately 1.9 million gallons (7,190 cubic meters) per day, 140,000 gallons (530 cubic meters) per day, and 170,000 gallons (640 cubic meters) per day, respectively. H Area has two domestic wells and three process production wells; L Area has two domestic wells. Until recently, two P-Area groundwater wells were used for domestic purpose. At present, these wells are not being used for domestic or process production. SRS is implementing a consolidation program for domestic wells. When this program is complete, DOE might take the domestic wells in C, F, H, and L Areas out of service or use them only for process water (Wells 1997).

3.2.2.3 SRS Hydrogeology

The aquifers of interest for C, F, H, L, and P Areas are the Upper Three Runs and Gordon Aquifers. The Upper Three Runs (water table) Aquifer is defined by the hydrogeologic properties of the Tinker/Santee Formation, the Dry Branch Formation, and the Tobacco Road Formation (DOE 1996a). Table 3.1-1 lists these formations.

The Gordon Confining Unit (green clay), which separates the Upper Three Runs and Gordon Aquifers, consists of the Warley Hill Formation and the Blue Bluff Member of the Santee Limestone (Table 3.1-1). It is not a continuous clay unit, but consists of several lenses of green and gray clay that thicken, thin, and pinch out abruptly. Locally, beds of calcareous mud add to the thickness of the unit with minor interbeds of clayey sand or sand. The vertical hydraulic conductivity ranges from 1.1×10^{-6} foot (3.4×10^{-5} centimeter) to 0.16 foot (4.9 centimeters) per day and the horizontal conductivity ranges from 5.4×10^{-6} foot (1.6×10^{-5} centimeter) to 5.7×10^{-3} foot (0.17 centimeter) per day (Aadland, Gellici, and Thayer 1995).

The Gordon Aquifer consists of the Congaree, Fourmile, and Snapp Formations. Table 3.1-1 lists the soil descriptions for these formations. The Gordon Aquifer is partially eroded near the Savannah River and Upper Three Runs. This aquifer is recharged directly by precipitation in

the outcrop area and at interstream drainage divides in and near the outcrop area, and by leakage from overlying and underlying aquifers. The northeast-to-southwest hydraulic gradient across SRS is consistent and averages 4.8 feet per mile (0.9 meter per kilometer). Based on pumping tests on 13 SRS wells, the average hydraulic conductivity is approximately 35 feet (10.7 meters) per day.

3.2.2.4 Groundwater Quality

Industrial solvents, metals, tritium, and other constituents used or generated on SRS have contaminated the shallow aquifers beneath 5 to 10 percent of the Site. In general, DOE does not use these aquifers for SRS operations or drinking water, although there are a few low-yield wells in the Gordon Aquifer. The shallow aquifer units discharge to SRS streams and eventually the Savannah River (Arnett and Mamatey 1997).

Most contaminated groundwater at SRS occurs beneath a few facilities; the contaminants reflect the operations and chemical processes performed at those facilities. At C Area, groundwater contaminants above regulatory or SRS guidelines include tritium and other radionuclides, bis (2-ethylhexyl) phthalate, carbon disulfide, lead, manganese, and chlorinated organics. At F and H Areas, contaminants above the guidelines include tritium and other radionuclides, metals, nitrates, sulfates, and chlorinated and volatile organics. At L Area, tritium, other radionuclides, carbon disulfide, chlorinated and volatile organics, and metals are in the groundwater at levels above the guidelines. Groundwater beneath the L-Area Disassembly Basin has been affected by metals, chlorinated organics, and tritium at levels above regulatory guidelines. Tables 3.2-4 through 3.2-8 list concentrations of individual analytes above regulatory or SRS guidelines for 1995 in C, F, H, L, and P Areas, respectively (WSRC 1995a). Figure 3.2-4 shows generalized groundwater contamination maximum values for analytes at or above regulatory or established SRS guidelines for the areas of concern.

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Table 3.2-4. C-Area maximum reported groundwater parameters in excess of regulatory and SRS limits.^a

Analyte	Concentration	Regulatory Limit
Aluminum ^b	6,430 µg/L	50 µg/L ^c
Bis (2-ethylhexyl) phthalate	23 µg/L	6 µg/L ^d
Iron ^b	10,500 µg/L	300 µg/L ^d
Lead ^b	301 µg/L	50 µg/L ^e
Manganese ^b	254 µg/L	50 µg/L ^c
Carbon disulfide	74 µg/L	10 µg/L ^f
Trichloroethylene	1,580 µg/L	5 µg/L ^d
Tetrachloroethylene	174 µg/L	5 µg/L ^d
Dichloromethane	8.7 µg/L	5 µg/L ^d
Total organic halogens	972 µg/L	50 µg/L ^f
Tritium	2.4×10^{-2} µCi/mL	2.0×10^{-5} µCi/mL ^d
Thallium	3.5 µg/L	2 µg/L ^d
Thorium-234	6.8×10^{-7} µCi/mL	4.01×10^{-7} µCi/mL ^g

a. µg/L = micrograms per liter; µCi/mL = microcuries per milliliter.

b. Total recoverable.

c. EPA National Secondary Drinking Water Standards (WSRC 1995a).

d. EPA Primary Drinking Water Standards (WSRC 1995a).

e. SCDHEC Final Primary Drinking Water Standards (WSRC 1995a).

f. Drinking Water Standards do not apply. Criterion 10 times a recently published 90th percentile detection limit was used (WSRC 1995a).

g. EPA Proposed Primary Drinking Water Standard (WSRC 1995a).

3.3 Air Resources

3.3.1 GENERAL METEOROLOGY

Based on data collected from SRS meteorological towers from 1987 through 1991 (the latest quality-assured 5-year data set), maximum wind direction frequencies at the Site are from the northeast and west-southwest and the average wind speed is 8.5 miles per hour (3.8 meters per second). The average annual temperature at the Site is 64°F (17.8°C). The atmosphere in the region is unstable approximately 56 percent of the time, neutral 23 percent of the time, and stable about 21 percent of the time (Shedrow 1993). In general, as the atmosphere becomes more unstable, atmospheric dispersion of airborne pollutants increases and ground-level pollutant concentrations decrease.

3.3.2 SEVERE WEATHER

The SRS area experiences an average of 55 thunderstorm days a year, 50 percent of which occur in June, July, and August (Shedrow 1993). On average, lightning strikes six times a year on a square-kilometer area (Hunter 1990). The highest windspeed recorded at Bush Field (Augusta, Georgia) between 1950 and 1993 was 62 miles (100 kilometers) per hour (NOAA 1994).

From 1954 to 1983, 37 reported tornadoes occurred in a 1-degree square of latitude and longitude that includes SRS (WSRC 1993). This frequency of occurrence is equivalent to an average of about one tornado per year. Tornado statistics indicate that the average frequency of a tornado striking any single point on the site is

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Table 3.2-5. F-Area maximum reported groundwater parameters in excess of regulatory and SRS limits.^a

Analyte	Concentration ($\mu\text{g/L}$ for metals and organics; $\mu\text{Ci/mL}$ for radioisotopes unless otherwise noted)	Regulatory limit ($\mu\text{g/L}$ for metals and organics; $\mu\text{Ci/mL}$ for radioisotopes)
Aluminum ^b	95,900	50 ^c
Beryllium ^b	10	4 ^d
Bis (2-ethylhexyl) phthalate	190	6 ^d
Cadmium ^b	243	5 ^d
Copper ^b	1,210	1,000 ^d
Chromium ^b	185	100 ^d
Iron ^b	261,000	300 ^d
Lead ^b	6,500	50 ^c
Lithium ^b	249	50 ^f
Manganese ^b	15,000	50 ^c
Mercury ^b	5.4	2 ^c
Nickel ^b	176	100 ^d
Carbon tetrachloride	23	5 ^d
Trichloroethylene	96	5 ^d
Trichlorofluoromethane	80	10 ^f
Tetrachloroethylene	42	5 ^d
Dichloromethane	65	5 ^d
1,2-dichloroethane	162	5 ^d
Total organic carbon	18,600	10,000
Total organic halogens	148	50 ^f
Nitrate as nitrogen	71,300	1,000 ^d
Nitrate-nitrite as nitrogen	384,000	10,000 ^d
Americium-241	9.9×10^{-8}	6.34×10^{-9g}
Cesium-137	4.4×10^{-7}	2.0×10^{-7h}
Cobalt ^b	665	40 ^f
Curium-243/244	1.6×10^{-7}	8.3×10^{-9g}
Curium-245/246	9.9×10^{-8}	6.23×10^{-9g}
Iodine-129	7.2×10^{-7}	1.0×10^{-9h}
Lithium ^b	56	50 ^f
Tritium	2.2×10^{-2}	2.0×10^{-5d}
Plutonium-238	2.3×10^{-8}	7.02×10^{-9g}
Radium-226	1.1×10^{-7}	$2.0 \times 10^{-8g,i}$
Radium-228	3.1×10^{-7}	$2.0 \times 10^{-8g,i}$
Nonvolatile beta	2.5×10^{-5}	5.0×10^{-8h}
Total alpha-emitting radium	1.6×10^{-7}	2.0×10^{-8g}
Gross alpha	2.5×10^{-6}	1.5×10^{-8d}
Strontium-89	7.1×10^{-7}	2.0×10^{-8h}
Strontium-90	7.4×10^{-6}	8.0×10^{-9d}
Thallium ^b	4.3	2.0 ^d
Thorium-234	9.5×10^{-7}	4.01×10^{-7g}
Uranium-233/234	4.8×10^{-7}	1.38×10^{-8g}
Uranium-235	5.0×10^{-8}	1.45×10^{-8g}
Uranium-238	1.3×10^{-6}	1.46×10^{-8g}

a. Abbreviations: $\mu\text{g/L}$ = micrograms per liter; $\mu\text{Ci/mL}$ = microcuries per milliliter.

b. Total recoverable.

c. EPA National Secondary Drinking Water Standard (WSRC 1995a).

d. EPA Final Primary Drinking Water Standard (WSRC 1995a).

e. SCDHEC Final Primary Drinking Water Standard (WSRC 1995a).

f. Drinking Water Standards do not apply. Criterion 10 times a recently published 90th percentile detection limit was used (WSRC 1995a).

g. EPA Proposed Primary Drinking Water Standard (WSRC 1995a).

h. EPA Interim Final Primary Drinking Water Standards (WSRC 1995a).

i. Radium-226/228 combined proposed Maximum Contaminant Level of 5.0×10^{-8} microcuries per milliliter.

Table 3.2-6. H-Area maximum reported groundwater parameters in excess of regulatory and SRS limits.^a

Analyte	Concentration ($\mu\text{g/L}$ for metals and organics; $\mu\text{Ci/mL}$ for radioisotopes)	Regulatory limit ($\mu\text{g/L}$ for metals and organics; $\mu\text{Ci/mL}$ for radioisotopes)
Aluminum ^b	2,800	50 ^c
Bis (2-ethylhexyl) phthalate	23	6 ^d
Iron ^b	7,990	300 ^d
Lead ^b	301	50 ^c
Manganese ^b	91	50 ^c
Trichloroethylene	1,580	50 ^c
Total Organic Halogens	972	50 ^d
Thallium ^b	4.0	2.0 ^d
Tritium	2.4×10^{-2}	2.0×10^{-5d}
Thorium-234	6.8×10^{-7}	4.01×10^{-7g}

a. Abbreviations: $\mu\text{g/L}$ = micrograms per liter; $\mu\text{Ci/mL}$ = microcuries per milliliter.

b. Total recoverable.

c. EPA National Secondary Drinking Water Standard (WSRC 1995a).

d. EPA Final Primary Drinking Standard (WSRC 1995a).

e. SCDHEC Final Primary Drinking Water Standard (WSRC 1995a).

f. Drinking Water Standards do not apply. Criterion 10 times a recently published 90th percentile detection limit was used (WSRC 1995a).

g. EPA Proposed Primary Drinking Water Standards (WSRC 1995a).

Table 3.2-7. L-Area maximum reported groundwater parameters in excess of regulatory and SRS limits.^a

Analyte	Concentration ($\mu\text{g/L}$ for metals and organics; $\mu\text{Ci/mL}$ for radioisotopes)	Regulatory limit ($\mu\text{g/L}$ for metals and organics; $\mu\text{Ci/mL}$ for radioisotopes)
Aluminum ^b	320	50 ^c
Boron ^b	1,590	300 ^d
Iron ^b	14,100	300 ^d
Lead ^b	58	50 ^c
Manganese ^b	771	50 ^c
Tetrachloroethylene	17	5 ^d
Total Organic Carbon	3.5×10^{-6}	10,000 ^f
Nitrate-nitrite as Nitrogen	268,000	10,000 ^d
Thallium ^b	7.4	2.0 ^d
Tritium	5.4×10^{-4}	2.0×10^{-5d}
Non-volatile Beta	1.7×10^{-6}	5.0×10^{-8g}

a. Abbreviations: $\mu\text{g/L}$ = micrograms per liter; $\mu\text{Ci/mL}$ = microcuries per milliliter.

b. Total recoverable.

c. EPA National Secondary Drinking Water Standard (WSRC 1995a).

d. EPA Final Primary Drinking Water Standard (WSRC 1995a).

e. SCDHEC Final Primary Drinking Water Standard (WSRC 1995a).

f. Drinking Water Standards do not apply. Criterion 10 times a recently published 90th percentile detection limit was used (WSRC 1995a).

g. EPA Interim Final Primary Drinking Water Standards (WSRC 1995a).

Table 3.2-8. P-Area maximum reported groundwater parameters in excess of regulatory and SRS limits.^a

Analyte	Concentration ($\mu\text{g/L}$ for metals and organics)	Regulatory limit ($\mu\text{g/L}$ for metals and organics)
Aluminum ^b	19,900	50 ^c
Iron ^b	22,200	300 ^d
Manganese ^b	419	50 ^c
Carbon tetrachloride	11	5 ^d
Trichloroethylene	24	50 ^d
Tetrachloroethylene	8.4	5 ^d
Total organic halogens	79	50 ^c
Tritium	$7.7 \times 10^{-2} \text{ Ci/mL}$	$2.0 \times 10^{-5d} \mu\text{Ci/mL}$
Strontium-90	$1.7 \times 10^{-6} \text{ Ci/mL}$	$8.0 \times 10^{-9d} \mu\text{Ci/mL}$

a. Abbreviations: $\mu\text{g/L}$ = micrograms per liter; $\mu\text{Ci/mL}$ = microcuries per milliliter.

b. Total recoverable.

c. EPA National Secondary Drinking Water Standard (WSRC 1995a).

d. EPA Final Primary Drinking Water Standard (WSRC 1995a).

e. Drinking Water Standards do not apply. Criterion 10 times a recently published 90th percentile detection limit was used (WSRC 1995a).

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2×10^{-4} per year or about once every 5,000 years (Weber et al. 1998). Since operations began in 1953, nine confirmed tornadoes have occurred on or near the Site. Nothing more than light damage occurred, with the exception of a tornado in October 1989 that caused considerable damage to forest resources in an undeveloped southeastern sector of the SRS (Shedrow 1993). From 1700 to 1992, 36 hurricanes crossed South Carolina, which resulted in a frequency of about one every 8 years (WSRC 1993). Because the SRS is about 100 miles (160 kilometers) inland, the winds associated with hurricanes have usually diminished below hurricane force [i.e., equal to or greater than a sustained wind speed of 75 miles per hour (33.5 meters per second)] before reaching the Site. Winds exceeding hurricane force have been observed only once at the SRS (Hurricane Gracie in 1959) (Shedrow 1993).

3.3.3 RADIOLOGICAL AIR QUALITY

DOE provides detailed summaries of radiological releases to the atmosphere from SRS operations, along with resulting concentrations and doses, in a series of annual environmental data reports. This section references several of those

documents, which contain additional information. The information enables comparisons of current data with potential releases, concentrations, and doses associated with each alternative.

In the SRS region, airborne radionuclides originate from natural sources (terrestrial and cosmic), worldwide fallout, and Site operations. DOE maintains a network of air monitoring stations on and around the Site to determine concentrations of radioactive particulates and aerosols in the air (Arnett and Mamatey 1998b).

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Table 3.3-1 lists average and maximum atmospheric radionuclide concentrations at the SRS boundary and at background monitoring locations [100-mile (160-kilometer) radius] during 1997. Tritium is the only radionuclide from the SRS detected routinely in offsite air samples above background (control) concentrations (Cummins, Martin, and Todd 1990, 1991; Arnett et al. 1992; Arnett, Karapatakis, and Mamatey 1993, 1994; Arnett and Mamatey 1996; Arnett and Mamatey 1997; Arnett and Mamatey 1998b). Table 3.3-2 lists 1997 radionuclide releases from each major operational group of SRS facilities. All radiological impacts are within regulatory requirements.

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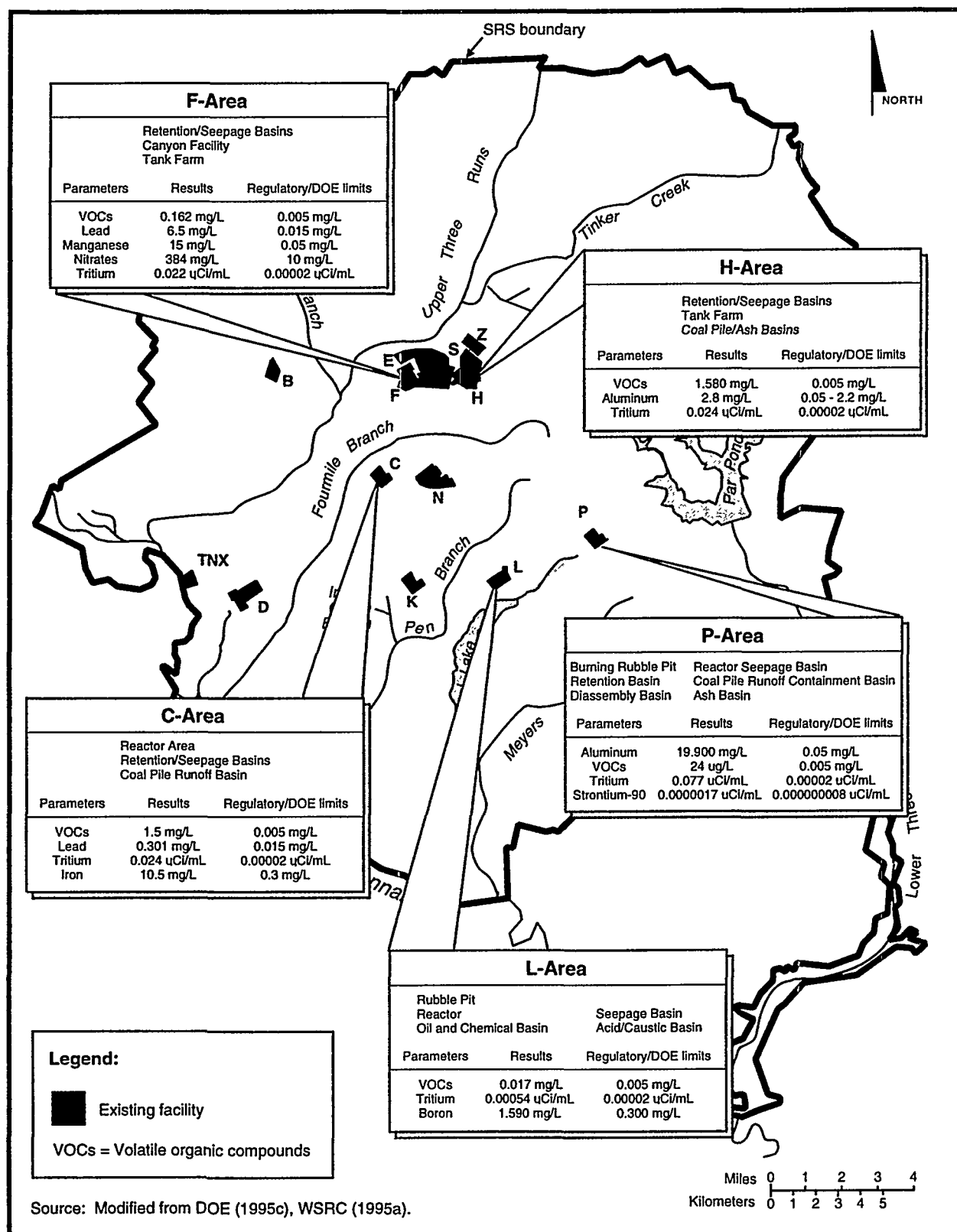


Figure 3.2-4. Maximum reported groundwater contamination at Savannah River Site.

Table 3.3-1. Radioactivity in air at SRS boundary and at 100-mile (160-kilometer) radius during 1997 (picocuries per cubic meter).^a

Location	Tritium	Gross alpha	Gross beta	Cobalt-60	Cesium-137	Strontium-89,90	Plutonium-238	Plutonium-239
Site boundary								
Average ^b	11	9.8×10 ⁻⁴	0.015	5.7×10 ⁻⁴	1.5×10 ⁻⁴	8.0×10 ⁻⁵	(c)	(c)
Maximum ^d	65	0.0033	0.032	0.024	0.0073	3.6×10 ⁻⁴	4.1×10 ⁻⁶	7.0×10 ⁻⁶
Background (100-mile radius)								
Average	3.2	0.0011	0.011	(c)	(c)	8.9×10 ⁻⁴	6.9×10 ⁻⁶	(c)
Maximum	5.4	0.0030	0.018	0.0073	0.0055	0.0019	4.2×10 ⁻⁵	2.6×10 ⁻⁵

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a. Source: Arnett and Mamatey (1998a).

b. The average value is the average value of the arithmetic means reported for the site perimeter sampling locations.

c. Below background levels.

d. The maximum value is the highest value of the maximums reported for the site perimeter sampling locations.

3.3.4 NONRADIOLOGICAL AIR QUALITY

The SRS is in the Augusta (Georgia) - Aiken (South Carolina) Interstate Air Quality Control Region. This region, which is designated a Class II area, is in compliance with National Ambient Air Quality Standards for criteria pollutants. Class II is the initial designation of any area that is not pristine; pristine areas include national parks or national wilderness areas. Criteria pollutants include sulfur dioxide, nitrogen oxides (reported as nitrogen dioxide), particulate matter (less than or equal to 10 microns in diameter), carbon monoxide, ozone, and lead (40 CFR 50).

DOE used the comprehensive emissions inventory data for 1996, which is the most recent available, to establish the baseline year for showing compliance with national and state air quality standards by calculating actual emission rates for existing sources of criteria pollutants. DOE based these emission rates on process knowledge, source testing, material balance, and EPA's Industrial Source Complex Air Dispersion Model.

SCDHEC has air quality regulatory authority over SRS. SCDHEC determines ambient air quality compliance based on air pollutant emissions and estimates of concentrations at the Site boundary based on atmospheric dispersion modeling. The SRS is in compliance with National Ambient Air Quality Standards for criteria pollutants and gaseous fluoride and with total suspended particulate standards, as required by SCDHEC Regulation R.61-62.5, Standard 2, "Ambient Air Quality Standards." Table 3.3-3 lists these standards and the results of the atmospheric dispersion modeling for baseline year 1996.

The SRS is in compliance with SCDHEC Regulation R.61-62.5, Standard 8, "Toxic Air Pollutants," which regulates the emission of 257 toxic air pollutants (WSRC 1994). DOE has identified emission sources for 139 of the 257 regulated air toxics; the modeled results indicate that the Site is in compliance with SCDHEC air quality standards. Table 3.3-4 lists toxic air pollutants that are the same as those the alternative actions described in this EIS would emit, and compares maximum downwind concentrations at the Site boundary for baseline year 1990, which is the most recent data available, to SCDHEC standards for toxic air pollutants.

Table 3.3-2. Radiological atmospheric releases by operational group for 1997.^a

Radionuclide ^b	Half-life	Reactors	Separations ^c	Reactor materials	Heavy water	SRTC ^d	Diffuse and fugitive ^e		Total
							Curies released		
Gases and vapors									
H-3 (oxide)	12.3 years	5.2×10 ³	3.3×10 ⁴		350		150		3.9×10 ⁴
H-3 (elem)	12.3 years		1.9×10 ⁴						1.9×10 ⁴
H-3 Total	12.3 years	5.2×10 ³	5.2×10 ⁴		350		150		5.8×10 ⁴
C-14	5.73×10 ³ years		3.1×10 ⁻²				1.9×10 ⁻⁸		3.1×10 ⁻²
Kr-85	10.73 years		9.6×10 ³						9.6×10 ³
I-129	1.57×10 ⁷ years		7.1×10 ⁻³				1.2×10 ⁻⁷		7.1×10 ⁻³
I-131	8.040 days		2.9×10 ⁻⁵			2.98×10 ⁻⁵			5.9×10 ⁻⁵
I-133	20.8 hours					4.92×10 ⁻⁴			4.9×10 ⁻⁴
Particulates									
Na-22	2.605 years						1.1×10 ⁻⁹		1.1×10 ⁻⁹
Mn-54	312.2 days						4.8×10 ⁻¹²		4.8×10 ⁻¹²
Co-57	271.8 days		2.2×10 ⁻⁷				1.0×10 ⁻⁹		2.1×10 ⁻⁷
Co-58	70.88 days						1.7×10 ⁻¹²		1.7×10 ⁻¹²
Co-60	5.271 years		3.5×10 ⁻⁷				9.1×10 ⁻⁷		1.3×10 ⁻⁶
Ni-59	7.6×10 ⁴ years						3.2×10 ⁻¹⁰		3.2×10 ⁻¹⁰
Ni-63	100 years						2.3×10 ⁻⁹		2.3×10 ⁻⁹
Zn-65	243.8 days						3.7×10 ⁻¹²		3.7×10 ⁻¹²
Se-79	6.5×10 ⁴ years						2.2×10 ⁻¹⁰		2.2×10 ⁻¹⁰
Sr-89,90 ^f	29.1 years	1.8×10 ⁻³	2.2×10 ⁻⁴	4.2×10 ⁻⁵	1.8×10 ⁻⁴		8.2×10 ⁻⁵		2.3×10 ⁻³
Zr-95	64.02 days						2.1×10 ⁻⁵		2.1×10 ⁻⁵
Nb-95	34.97 days						1.6×10 ⁻¹⁵		1.6×10 ⁻¹⁵
Tc-99	2.13×10 ⁵ years						3.6×10 ⁻⁸		3.6×10 ⁻⁸
Ru-106	1.020 years						0.070		0.070
Sn-126	1×10 ⁵ years						3.4×10 ⁻¹⁵		3.4×10 ⁻¹⁵
Sb-124	60.2 days						3.4×10 ⁻¹²		3.4×10 ⁻¹²
Sb-125	2.758 years						5.9×10 ⁻⁷		5.9×10 ⁻⁷
Cs-134	2.065 years		1.4×10 ⁻⁶				1.2×10 ⁻⁹		1.4×10 ⁻⁶
Cs-137	30.17 years	2.5×10 ⁻⁴	4.2×10 ⁻⁴		2.9×10 ⁻⁶		4.2×10 ⁻³		4.9×10 ⁻³
Ba-133	10.53 years						3.0×10 ⁻¹²		3.0×10 ⁻¹²
Ce-144	284.6 days		4.2×10 ⁻⁶				6.1×10 ⁻⁶		1.0×10 ⁻⁵
Pm-144	360 days						1.3×10 ⁻¹²		1.3×10 ⁻¹²

TC

Table 3.3-2. (Continued).

Radionuclide ^b	Half-life	Reactors	Separations ^c	Reactor materials	Heavy water	SRTC ^d	Diffuse and fugitive ^e	Total	TC
Particulates (continued)									
Pm-147	2.6234 years						1.0×10 ⁻⁸	1.0×10 ⁻⁸	
Eu-152	13.48 years						5.3×10 ⁻⁹	5.3×10 ⁻⁹	
Eu-154	8.59 years		1.5×10 ⁻⁷				6.4×10 ⁻⁶	6.6×10 ⁻⁶	
Eu-155	4.71 years		4.9×10 ⁻⁶				1.7×10 ⁻⁶	6.6×10 ⁻⁶	
Ra-226	1.6×10 ³ years						1.2×10 ⁻⁸	1.2×10 ⁻⁸	
Ra-228	5.76 years						1.8×10 ⁻¹⁰	1.8×10 ⁻¹⁰	
Th-228	1.913 years						2.2×10 ⁻¹⁰	2.2×10 ⁻¹⁰	
Th-230	7.54×10 ⁴ years						2.0×10 ⁻¹⁰	2.0×10 ⁻¹⁰	
Th-232	1.40×10 ¹⁰ years						1.4×10 ⁻¹⁰	1.4×10 ⁻¹⁰	
Th-234	24.10 days						2.3×10 ⁻¹⁰	2.3×10 ⁻¹⁰	
Pa-231	3.28×10 ⁴ years						1.0×10 ⁻⁹	1.0×10 ⁻⁹	
Pa-234	6.69 hours						2.3×10 ⁻¹⁰	2.3×10 ⁻¹⁰	
U-233	1.592×10 ⁵ years						2.1×10 ⁻⁸	2.1×10 ⁻⁸	
U-234	2.46×10 ⁵ years		8.0×10 ⁻⁶	4.0×10 ⁻⁶			1.5×10 ⁻⁵	2.7×10 ⁻⁵	
U-235	7.04×10 ⁸ years		6.3×10 ⁻⁷	6.4×10 ⁻⁷			4.8×10 ⁻⁷	1.8×10 ⁻⁶	
U-236	2.342×10 ⁷ years						4.8×10 ⁻⁷	4.8×10 ⁻⁷	
U-238	4.47×10 ⁹ years		1.9×10 ⁻⁵	1.7×10 ⁻⁶			3.5×10 ⁻⁵	5.6×10 ⁻⁵	
Np-237	2.14×10 ⁶ years						1.4×10 ⁻⁹	1.4×10 ⁻⁹	
Np-239	2.35 days						2.2×10 ⁻⁷	2.2×10 ⁻⁷	
Pu-238	87.7 years		3.3×10 ⁻⁵	4.4×10 ⁻⁹			3.6×10 ⁻⁴	3.9×10 ⁻⁴	
Pu-239 ^g	2.410×10 ⁴ years	2.9×10 ⁻⁴	5.1×10 ⁻⁵	6.9×10 ⁻⁶	2.3×10 ⁻⁵	2.5×10 ⁻⁶	6.9×10 ⁻⁶	3.8×10 ⁻⁴	
Pu-240	6.56×10 ³ years						1.1×10 ⁻⁶	1.1×10 ⁻⁶	
Pu-241	14.4 years						5.2×10 ⁻⁵	5.2×10 ⁻⁵	
Pu-242	3.75×10 ⁵ years						3.7×10 ⁻¹¹	3.7×10 ⁻¹¹	
Am-241	432.7 years		1.4×10 ⁻⁵	1.2×10 ⁻⁸			8.7×10 ⁻⁷	1.5×10 ⁻⁵	
Am-243	7.37×10 ³ yr						1.8×10 ⁻⁵	1.8×10 ⁻⁵	
Cm-242	162.8 days						8.2×10 ⁻¹²	8.2×10 ⁻¹²	

Table 3.3-2. (Continued).

Radionuclide ^b	Half-life	Reactors	Separations ^c	Reactor materials	Heavy water	SRTC ^d	Diffuse and fugitive ^e	Total
Particulates (continued)								
Cm-244	18.1 years		2.5×10 ⁻³	2.0×10 ⁻¹⁰			1.3×10 ⁻⁴	1.5×10 ⁻⁴
Cm-245	8.5×10 ³ years						1.9×10 ⁻¹²	1.9×10 ⁻¹²
a. Source: Amett and Mamatey (1998a). b. H = hydrogen (H-3 = tritium), C = carbon, Kr = krypton, I = iodine, Na = sodium, Mn = manganese, Co = cobalt, Ni = nickel, Zn = zinc, Se = selenium, Sr = strontium, Zr = zirconium, Nb = niobium, Tc = technetium, Ru = ruthenium, Sn = tin, Sb = antimony, Cs = cesium, Ba = barium, Ce = cerium, Pm = promethium, Eu = europium, Ra = radium, Th = thorium, Pa = protactinium, U = uranium, Np = neptunium, Pu = plutonium, Am = americium, Cm = curium. c. Includes F- and H-Area releases. d. SRTC = Savannah River Technology Center. e. Estimated releases from minor unmonitored diffuse and fugitive sources. f. Includes unidentified beta emissions. g. Includes unidentified alpha emissions.								
								TC

Table 3.3-3. SRS baseline air quality for maximum potential emissions and observed ambient concentrations.

Pollutant	Averaging time	SCDHEC ambient standard ($\mu\text{g}/\text{m}^3$) ^a	Estimated SRS baseline concentration ($\mu\text{g}/\text{m}^3$) ^b
Criteria pollutants			
Sulfur dioxide (as SO_x) ^c	3-hr	1,300	1,200
	24-hr	365	350
	Annual	80	34
Total suspended particulates	Annual	75	67
Particulate matter ($\leq 10 \mu\text{m}$) ^d	24-hr	150	133
	Annual	50	25
Carbon monoxide	1-hr	40,000	10,000
	8-hr	10,000	6,900
Nitrogen dioxides (as NO_x) ^e	Annual	100	26
Lead	Calendar Quarterly mean	1.5	0.03
Ozone (as total VOCs) ^f	1-hr	235	NA ^g
Toxic/hazardous air pollutants			
Benzene	24-hr	150	3.9
Beryllium	24-hr	0.01	0.009
Biphenyl	24-hr	6	0.02
Mercury	24-hr	0.25	0.03
Methyl alcohol (methanol)	24-hr	1,310	0.9

TC

SO_x = oxides of sulfur; NO_x = oxides of nitrogen; VOCs = volatile organic compounds; NA = not available.

- Source: SCDHEC Standard 2, "Ambient Air Quality Standards," and Standard 8, "Toxic Air Pollutants" (SCDHEC 1976).
- Source: Hunter (1999). Concentration is the sum of modeled air concentrations using the permitted maximum potential emissions from the 1998 air emissions inventory for all SRS sources not exempted by Clean Air Act Title V requirements and observed concentrations from nearby ambient air monitoring stations.
- Based on emissions for all oxides of sulfur (SO_x).
- New NAAQS for particulate matter ≤ 2.5 microns (24-hour limit of $65 \mu\text{g}/\text{m}^3$ and an annual average limit of $15 \mu\text{g}/\text{m}^3$) will become enforceable during the life of this project.
- Based on emissions for all oxides of nitrogen (NO_x).
- New NAAQS for ozone (8 hours limit of 0.08 parts per million) will become enforceable during the life of this project.
- Ambient concentrations of VOCs, which are precursors to ozone, can be used to provide a highly conservative bounding estimate for ozone but should not be used for explicit assessments of compliance with the ozone standard. Not all the VOCs emitted will result in the formation of ozone, and there is no method to directly correlate the two quantities. For purposes of estimating ozone concentrations from all SRS operations, no value for total VOCs is provided since the estimate would be overly conservative.

Table 3.3-4. Estimated 24-hour average ambient concentrations at SRS boundary - toxic air pollutants regulated by South Carolina from SRS sources.^a

Pollutant ^b	Concentration ($\mu\text{g}/\text{m}^3$) ^c	Regulatory standard ($\mu\text{g}/\text{m}^3$)	Concentration as a per- cent of standard (%)
Benzene	31	150	20.70
Hexane	0.07	200	0.04
Nitric acid	6.70	125	5.40
Sodium hydroxide	0.01	20	0.05
Toluene	1.60	2,000	0.08
Xylene	3.80	4,350	0.09

 $\mu\text{g}/\text{m}^3$ = micrograms per cubic meter.

a. Source: WSRC (1994).

b. Pollutants listed include air toxics of interest in relation to spent nuclear fuel management alternatives. (Section 5.2 addresses the effects of all air toxics.)

c. Based on actual emissions from existing SRS sources plus maximum potential emissions for sources permitted for construction through December 1992.

DOE measures nonradiological air emissions from SRS facilities at their points of discharge by direct measurement, sample extraction and measurement, or calculation of the emissions using process knowledge. Using monitoring data and meteorological information, DOE estimates the concentration of certain pollutants at the Site boundary. The Site is in compliance with National Ambient Air Quality Standards.

TC | The Environmental Protection Agency approved revisions to the national ambient air quality standards for ozone and particulate matter that became effective on September 16, 1997. However, on May 14, 1999, in response to challenges filed by industry and others, the U.S. Court of Appeals for the District of Columbia Circuit issued a split opinion (2 to 1) directing EPA to develop a new particulate matter standard (meanwhile reverting back to the previous PM_{10} standard) and ruling that the new ozone standard "cannot be enforced" (EPA 1999). The EPA has asked the U.S. Department of Justice to appeal this decision and take all judicial steps necessary to overturn the decision. Therefore, it is uncertain at this time when new ozone and particulate matter standards will become enforceable.

approximately two-thirds forested and one-third cropland and pastures. An extensive forest management program conducted by the Savannah River Natural Resources Management and Research Institute (SRI), which is part of the U.S. Forest Service, has converted many croplands and pastures to pine plantations. At present, more than 90 percent of the SRS is forested.

The Site provides more than 181,000 acres (734 square kilometers) of contiguous forested cover broken only by unpaved secondary roads, transmission line corridors in various stages of succession, a few paved primary roads, and scattered industrial facilities. Carolina bays, the Savannah River Swamp, and several relatively intact longleaf pine-wiregrass communities contribute to the biodiversity of the SRS and the entire region.

Under some of the alternatives described in Chapter 2, DOE proposes to construct and operate a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility at SRS to receive, characterize, condition, treat, package, and dry-store spent nuclear fuel before shipping it to a geologic repository. If not located in an existing reactor building, the site for either of these facilities would cover approximately 15 acres (0.061 square kilometer), including the

3.4 Ecological Resources

The U.S. Government acquired the land that became SRS in 1951. At that time, the Site was

building footprint(s), construction area needs, and security requirements (WSRC 1996a).

As described in Chapter 2, this Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would be in L Area (preferred site), C Area, or P Area. Facilities to implement the New Processing Technology Alternative also could be located inside a reactor building, such as Building 105-L.

The proposed site for any new facility in L Area is a ridge that runs southwest-to-northeast approximately 0.5 mile (0.8 kilometer) from the Steel Creek floodplain. The site, which is wholly within the developed portion of L Area, is bounded by L Reactor to the west, a rail spur (L Line) to the north, and paved access roads to the east and south. The area consists of buildings, paved areas, graveled areas, and mowed turf grasses. The site is inside 6-foot (1.8-meter) security fences and has negligible value as wildlife habitat.

An upland pine stand is immediately east of the proposed site, adjacent to the fenced area. The stand is primarily slash pines (*Pinus elliotti*) that the Forest Service planted in the mid-1950s, with small areas of long-leaf (*P. palustris*) and loblolly pine (*P. taeda*) planted in the 1940s (SRFS 1997). Understory species include black cherry (*Prunus serotina*), wax myrtle (*Myrica cerifera*) and yellow jessamine (*Gelsemium sempervirens*). SRI manages forested areas such as this for timber production and wildlife.

Wildlife characteristically found in SRS pine plantations include toads (i.e., the southern toad, [*Bufo terrestris*]), lizards (e.g., the eastern fence lizard, [*Sceloporus undulatus*]), snakes (e.g., the black racer, [*Coluber constrictor*]), songbirds (e.g., the brown-headed nuthatch [*Sitta pusilla*], and the pine warbler [*Dendroica pinus*]), birds of prey (e.g., the sharp-shinned hawk [*Accipiter striatus*]), and a number of mammal species (e.g., the cotton mouse [*Peromyscus gossypinus*]), the gray squirrel [*Sciurus carolinensis*], the opossum [*Didelphis virginiana*], and the white-tailed deer [*Odocoileus virginianus*]) (Sprunt and Chamberlain 1970; Cothran et

al. 1991; Gibbons and Semlitsch 1991; Halverson et al. 1997).

The proposed site for a new facility in C Area is on a plateau that rises between the floodplains of Fourmile Branch to the north and Castor Creek to the south. The entire site is inside the developed portion of C Area, surrounded by security fencing. The area consists of buildings, paved areas, graveled areas, and mowed turf grasses. A paved access road, a railroad spur, and two transmission lines cross the site. It provides little or no wildlife habitat. The areas immediately north and south of the site are forested, primarily with long-leaf and loblolly pine planted in the 1950s. The shrub layer contains young oaks (*Quercus* spp.) black cherry, hawthorne (*Crataegus* sp.), wax myrtle, and bear-grass (*Yucca filamentosa*). The wildlife species listed for L Area occur in these woods as well.

The proposed facility site in P Area is a broad hilltop above the headwaters of Steel Creek (to the west), Meyers Branch (to the south), and Lower Three Runs/Par Pond (to the east). The western two-thirds of the area (adjacent to the P-Area fence) is meadow-like, comprised mostly of lawn grasses and a few common forbs, such as low hop clover (*Trifolium dubium*) and smooth vetch (*Vicia dasycarpa*). The remainder of the area is wooded, with trees that appear to have regenerated since P Area was developed in the early 1950s. The canopy layer is dominated by laurel oak (*Quercus laurifolia*), water oak (*Q. nigra*), blackjack oak (*Q. marilandica*), mockernut hickory (*Carya alba*), and long-leaf pine. In the sub-canopy and shrub layer, species such as *Q. laevis* (turkey oak), huckleberry (*Vaccinium stamineum*), and hawthorne are well represented. Wooded areas to the north and east of the site are predominantly slash pines that were planted in the 1950s and loblolly pines that were planted in the 1980s (SRFS 1997). Because it is regularly mowed, the grassy area provides limited wildlife habitat. The wooded areas presumably provide habitat for many of the wildlife species mentioned above.

Under the Endangered Species Act of 1973 the Federal government provides protection to six species that occur on the SRS: American alligator (*Alligator mississippiensis*; threatened due to similarity of appearance to the endangered American crocodile), short-nosed sturgeon (*Acipenser brevirostrum*; endangered), bald eagle (*Haliaeetus leucocephalus*; threatened), wood stork (*Mycteria americana*; endangered), red-cockaded woodpecker (*Picoides borealis*; endangered), and smooth purple coneflower (*Echinacea laevigata*; endangered) (SRFS 1994). None of these species is known to occur on or near the proposed facility sites in L, C, P, F, or H Areas, which are located on previously disturbed areas (SRFS 1996).

3.5 Socioeconomics

Approximately 90 percent of the 1995 SRS workforce lived in the SRS region of influence which includes Aiken, Allendale, Bamberg, and Barnwell Counties in South Carolina, and Columbia and Richmond Counties in Georgia. *Socioeconomic Characteristics of Selected Counties and Communities Adjacent to the Savannah River Site* (HNUS 1997) contains additional information on the economic and demographic characteristics of the six-county region.

3.5.1 EMPLOYMENT

Between 1980 and 1990, total employment in the six-county region increased from 181,072 to 241,409, an average annual growth rate of approximately 2.9 percent. The unemployment rates for 1980 and 1990 were 7.3 percent and 4.7 percent, respectively (HNUS 1997). In 1994, regional employment was 243,854, an increase of only 1 percent since 1990. Over the next 10-year period, employment in the region is projected to increase at an average rate of slightly less than 1 percent per year, reaching approximately 264,000 by 2004 (HNUS 1997).

The increase in employment in the 1980s was spurred in part by the buildup in employment at

the SRS during the middle and late years of the decade, and in part by the improved national economy. The flat increases in regional employment since 1990 are the result of the mild national recession from 1990 to 1992, followed by the decreases in SRS employment, discussed below.

At the beginning of fiscal year 1996, employment at SRS was 16,625, approximately 7 percent of regional employment, with an associated annual payroll of approximately \$634 million. This represents a decrease of 6,726 in SRS employment since 1992 and an associated payroll reduction of \$466 million from more than \$1.1 billion. Site employment declined through attrition by approximately 950 jobs between the fall of 1995 and the fall of 1996 and by another approximately 850 jobs in early 1997 through involuntary separations. By March 1998, the SRS workforce was reported at 14,014 persons (DOE 1998).

3.5.2 POPULATION

Based on state and Federal agency surveys and trends, the estimated 1998 population in the region of influence was 466,222. About 90 percent lived in Aiken (29 percent), Columbia (20 percent), and Richmond (41 percent) counties. The population in the region grew at an annual growth rate of about 6.5 percent between 1990 and 1998 (U.S. Bureau of the Census 1999). Columbia County, and to a lesser extent Aiken County, contributed to most of the growth due to in-migration from other region of influence counties and other states. Over the same period Bamberg and Barnwell counties experienced net out-migration. In 2000, the population in the six-county region is expected to be approximately 498,900. Over the next 10-year period, the regional population should grow at a projected rate of 1 to 2 percent per year, reaching approximately 533,400 by 2010. According to census data, in 1990 the estimated average number of persons per household in the six-county region was 2.72, and the median age of the population was 31.8 years (HNUS 1997).

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3.5.3 COMMUNITY CHARACTERISTICS

Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations* (February 11, 1994), directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse human health or environmental effects of their programs, policies, and activities on minority and low-income populations. Executive Order 12898 also directs the Administrator of EPA to convene an interagency Federal Working Group on Environmental Justice.

The Working Group has provided guidance to Federal agencies on criteria for identifying disproportionately high and adverse human health or environmental effects on minority and low-income populations (EPA 1998). In addition, the Council on Environmental Quality, in consultation with EPA and other Federal agencies, has developed guidance for identifying and addressing environmental justice concerns during the National Environmental Policy Act (NEPA) process (CEQ 1998). DOE has based the environmental justice analysis in this document on those guidance documents. Further, in coordination with the Working Group, DOE is developing internal guidance for implementing the Executive Order.

Potential offsite health impacts from the proposed action would result from releases to the air and to the Savannah River downstream of the SRS. For air releases, DOE performed standard population dose analyses on a 50-mile (80-kilometer) radius because reasonably foreseeable dose levels beyond that distance would be negligible. For liquid releases, the region of interest includes areas that draw drinking water from the river (Beaufort and Jasper Counties in South Carolina and Effingham and Chatham Counties in Georgia).

The analysis included data (U.S. Bureau of the Census 1990a,b) for populations in census tracts with at least 20 percent of their area in the 50-mile radius and all tracts from Beaufort and Jasper Counties and Effingham and Chatham

Counties, which are downstream of the Site. DOE used data from each census tract in this combined region to identify the racial composition of communities and the number of persons characterized by the U.S. Bureau of the Census as living in poverty. The combined region contains 247 census tracts, 99 in South Carolina and 148 in Georgia.

Tables 3.5-1 and 3.5-2 list racial and poverty characteristics, respectively, of the population in the combined region. Table 3.5-1 indicates a total population of more than 993,000 in the area. Of that population, approximately 618,000 (62.2 percent) are white. In the minority population, approximately 94 percent are African American; the remainder are small percentages of Asian, Hispanic, and Native American persons. Figure 3.5-1 shows the distribution of minorities by census tract areas in the SRS region.

Executive Order 12898 does not define minority populations. One approach to a definition is to identify communities that contain a simple majority of minorities (greater than or equal to 50 percent of the total community population). A second approach, proposed by EPA for environmental justice purposes, defines minority communities as those that have higher-than-average (over the region of interest) percentages of minority persons (EPA 1994). The shading patterns in Figure 3.5-1 indicate census tracts where (1) minorities constitute 50 percent or more of the total population, or (2) minorities constitute between 35 percent and 50 percent of the total population. For this analysis, DOE has adopted the second, more expansive, approach to identify minority communities.

The combined region has 80 tracts (32.4 percent) where minority populations constitute 50 percent or more of the total population. In an additional 50 tracts (13.5 percent), minorities constitute between 35 and 50 percent of the population. These tracts are distributed throughout the region, although there are more toward the south and in the immediate vicinities of Augusta and Savannah, Georgia.

Table 3.5-1. General racial characteristics of population in SRS region of interest.^a

State	Total population	White	African American	Hispanic	Asian	Native American	Other	Minority	Percent minority ^b
South Carolina	418,685	267,639	144,147	3,899	1,734	911	355	151,046	36.08
Georgia	<u>574,982</u>	<u>350,233</u>	<u>208,017</u>	<u>7,245</u>	<u>7,463</u>	<u>1,546</u>	<u>478</u>	<u>224,749</u>	<u>39.09</u>
Total	993,667	617,872	352,164	11,144	9,197	2,457	833	375,795	37.82

a. Source: U.S. Bureau of the Census (1990a).

b. People of color population divided by total population.

Table 3.5-2. General poverty characteristics of population in SRS region of interest.^a

Area	Total population	Persons living in poverty ^b	Percent living in poverty
South Carolina	418,685	72,345	17.28
Georgia	<u>574,982</u>	<u>96,672</u>	<u>16.81</u>
Total	993,667	169,017	17.01

a. Source: U.S. Bureau of the Census (1990b).

b. Families with income less than the statistical poverty threshold, which in 1990 was 1989 income of \$8,076 for a family of two.

Low-income communities are those in which 25 percent or more of the population is characterized as living in poverty (EPA 1993). The U.S. Bureau of the Census defines persons in poverty as those whose income is less than a "statistical poverty threshold." This threshold is a weighted average based on family size and the age of the persons in the family. The baseline threshold for the 1990 census was a 1989 income of \$8,076 for a family of two.

Table 3.5-2 indicates that in the SRS region, more than 169,000 persons (17 percent of the population) are characterized as living in poverty. In Figure 3.5-2, shaded census tracts identify low-income communities. In the region, 72 tracts (29.1 percent) are low-income communities. These tracts are distributed throughout the region of analysis, but primarily to the south and west of SRS.

3.6 Cultural Resources

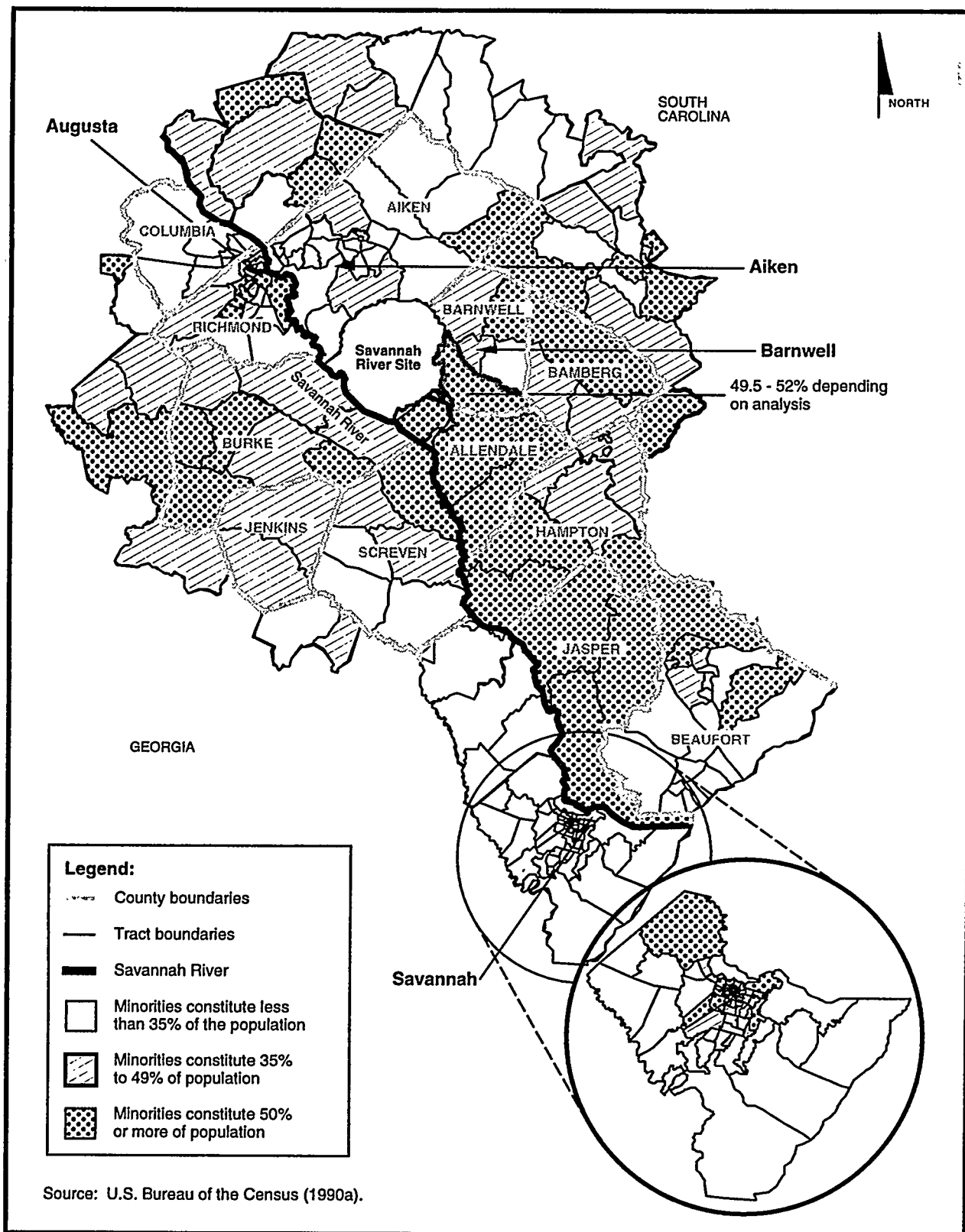
Through a cooperative agreement, DOE and the South Carolina Institute of Archaeology and Anthropology, University of South Carolina, conduct the Savannah River Archaeological Re-

search Program to provide on the SRS services required by Federal law for the protection and management of archaeological resources. Ongoing research programs work in conjunction with the South Carolina State Historic Preservation Officer. They provide theoretical, methodological, and empirical bases for assessing site significance using the compliance process specified by law. Archaeological investigations usually begin through the Site Use Program, which requires a permit for clearing land on the SRS.

The archaeological research has provided considerable information about the distribution and content of archaeological and historic sites on SRS. Savannah River archaeologists have examined SRS land since 1974. To date they have examined 60 percent of the 300-square-mile (800-square kilometer) area and recorded more than 1,200 archaeological sites (HNUS 1997). Most (approximately 75 percent) of these sites are prehistoric.

The activities associated with the proposed action and alternatives for spent nuclear fuel management at SRS that could affect cultural

EC



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Figure 3.5-1. Distribution of minorities by census tracts in the SRS region of analysis.

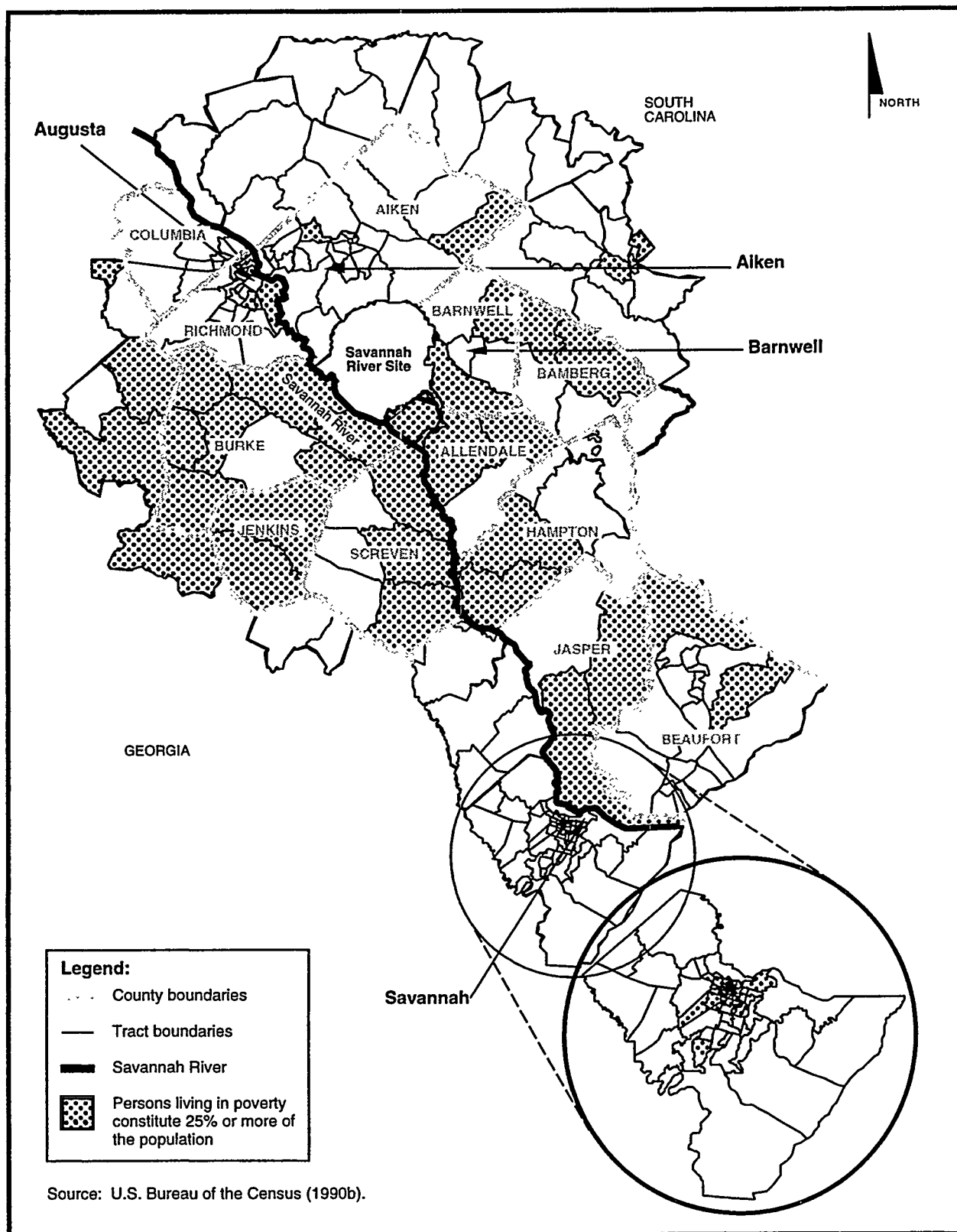


Figure 3.5-2. Low income census tracts in the SRS region of analysis.

resources are the use of one of the three sites for the proposed Transfer and Storage Facility or Transfer, Storage, and Treatment Facility.

The sites are in reactor areas (L, C, and P) within 100 to 400 yards (91 to 366 meters) of the reactor buildings. The Savannah River Archaeological Research Program has not examined any areas in and immediately around the reactors. Construction of these facilities took place before the enactment of Federal regulations to protect historic resources. Archaeological resources in the footprints of the three preferred sites would be unlikely to have survived reactor construction, although 1951 aerial photographs show that the C- and L-Area sites had homelands before the development of the SRS in the early 1950s (Sassaman 1997a,b).

The potential for prehistoric sites in the preferred locations is limited. The P-Area site is in archaeology site density Zone 2, which has moderate potential for prehistoric archaeological sites of significance. The L-Area site is in archaeological site density Zone 3, which has the least potential for prehistoric sites of significance. C Area is divided between Zones 2 and 3. However, in all cases, reactor construction activities probably destroyed or severely damaged any prehistoric deposits (Sassaman 1997a,b).

3.7 Public and Worker Health

3.7.1 PUBLIC RADIOLOGICAL HEALTH

Because there are many sources of radiation in the human environment, evaluations of radioactive releases from nuclear facilities must consider all ionizing radiation to which people are routinely exposed.

Doses of radiation are expressed as millirem (mrem), rem (1,000 millirem), and person-rem (which is the average individual doses times the population).

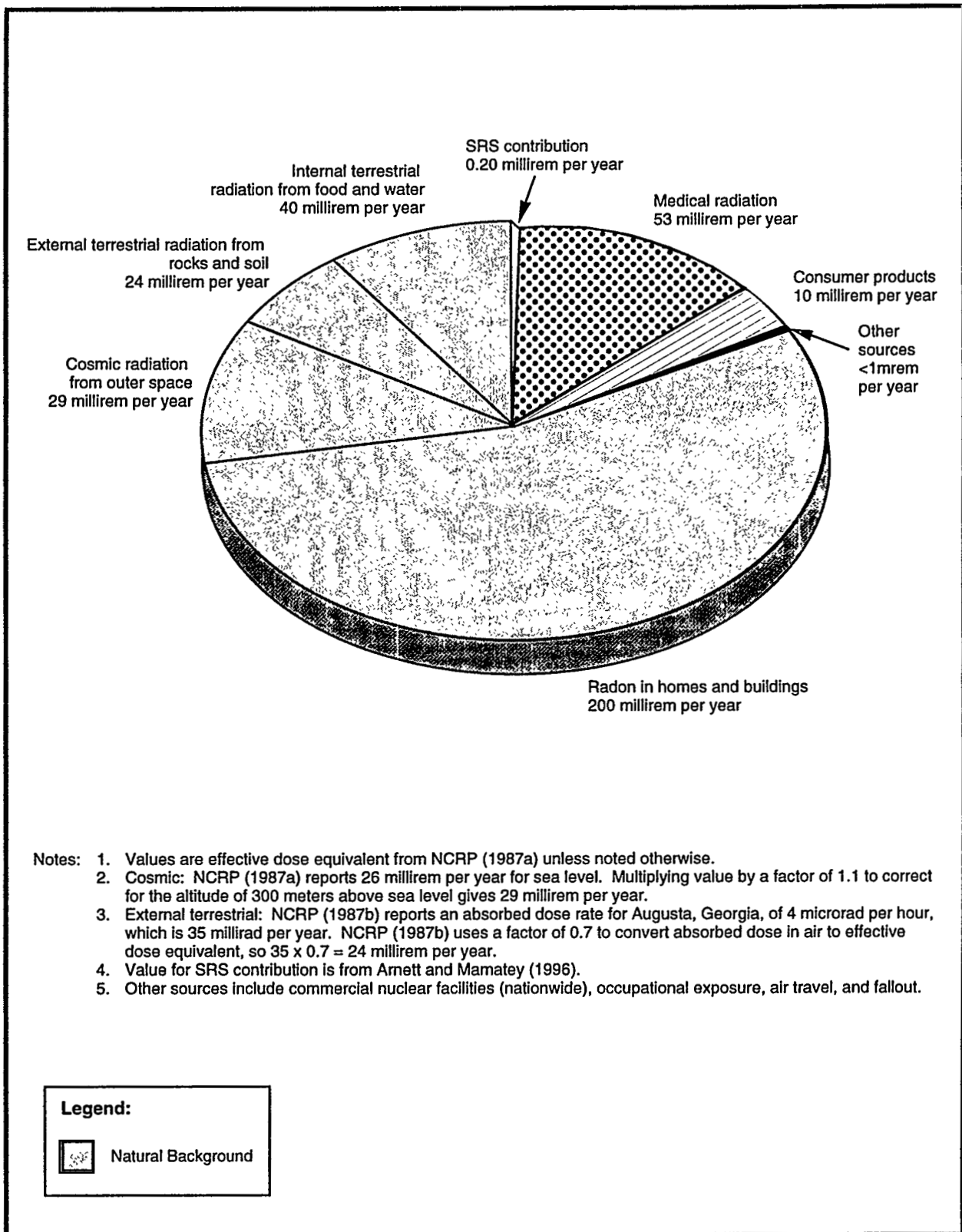
An individual's radiation exposure in the vicinity of SRS amounts to approximately 357 millirem per year, which is comprised of

natural background radiation from cosmic, terrestrial, and internal body sources, radiation from medical diagnostic and therapeutic practices, weapons test fallout, consumer and industrial products, and nuclear facilities. Figure 3.7-1 shows the relative contributions of each source to people living near SRS. All radiation doses mentioned in this EIS are effective dose equivalents; internal exposures are committed effective dose equivalents.

Releases of radioactivity to the environment from SRS account for less than 0.1 percent of the total annual average environmental radiation dose to individuals within 50 miles (80 kilometers) of the Site. Natural background radiation contributes about 293 millirem per year, or 82 percent of the annual dose of 357 millirem received by an average member of the population within 50 miles of the Site. Based on national averages, medical exposure accounts for an additional 14.8 percent of the annual dose, and combined doses from weapons test fallout, consumer and industrial products, and air travel account for about 3 percent (NCRP 1987a).

Other nuclear facilities within 50 miles (80 kilometers) of SRS include a low-level waste disposal site operated by Chem-Nuclear Systems, Inc., near the eastern Site boundary and Georgia Power Company's Vogtle Electric Generating Plant, directly across the Savannah River from the Site. In addition, Carolina Metals, Inc., which is northwest of Boiling Springs in Barnwell County, processes depleted uranium.

South Carolina Nuclear Facility Monitoring - Annual Report 1992 (SCDHEC 1992) documents that the Chem-Nuclear and Carolina Metals facilities do not influence radioactivity levels in the air, precipitation, groundwater, soil, or vegetation. Plant Vogtle began commercial operation in 1987: 1992 releases produced an annual dose of 0.17 millirem to the maximally exposed individual at the plant boundary and a total population dose within a 50-mile (80-kilometer) radius of 0.057 person-rem (NRC 1996).



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Figure 3.7-1. Major sources of radiation exposure in the vicinity of the Savannah River Site.

TC | In 1997, releases of radioactive material to the environment from SRS operations resulted in a maximum individual dose of 0.05 millirem per year in the west-southwest sector of the Site boundary from atmospheric releases, and a maximum dose from liquid releases of 0.13 millirem per year, for a maximum total annual dose at the boundary of 0.18 millirem (Arnett and Mamatey 1998b). The maximum dose to downstream consumers of Savannah River water – 0.07 millirem per year – occurred to users of the Port Wentworth and the Beaufort-Jasper public water supplies (Arnett and Mamatey 1998b).

TC | In 1990 the population within 50 miles (80 kilometers) of the Site was approximately 620,100. The collective effective dose equivalent to that population in 1997 was 2.2 person-rem from atmospheric releases. The 1990 population of 65,000 people using water from the Cherokee Hill Water Treatment Plant near Port Wentworth, Georgia, and the Beaufort-Jasper Water Treatment Plant near Beaufort, South Carolina, received a collective dose equivalent of 2.4 person-rem in 1997 (Arnett and Mamatey 1998b). Population statistics indicate that cancer caused 23.2 percent of the deaths in the United States in 1994 (CDC 1998). If this percentage of deaths from cancer continues, 23.2 percent of the U.S. population will contract a fatal cancer from all causes. Thus, in the population of 620,100 within 50 miles of SRS, 143,863 persons will be likely to contract fatal cancers from all causes. The total population dose from SRS of 4.6 person-rem (2.2 person-rem from atmospheric pathways plus 2.4 person-rem from water pathways) could result in 0.0023 additional latent cancer death in the same population [based on 0.0005 cancer death per person-rem (NCRP 1993)].

3.7.2 PUBLIC NONRADIOLOGICAL HEALTH

The hazards associated with the alternatives described in this EIS include exposure to nonradiological chemicals in the form of water and air

pollution (see Sections 3.2 and 3.3). Table 3.3-3 lists ambient air quality standards and concentrations for selected pollutants. The purpose of these standards is to protect the public health and welfare. The concentrations of pollutants from SRS sources, listed in Table 3.3-2, are lower than the standards. Section 3.2 discusses water quality in the SRS vicinity.

3.7.3 WORKER RADIOLOGICAL HEALTH

One of the major goals of the SRS Health Protection Program is to keep worker exposures to radiation and radioactive material as low as reasonably achievable (ALARA). Such a program must evaluate both external and internal exposures with the goal to minimize the total effective dose equivalent. An effective ALARA program must also balance minimizing individual worker doses with minimizing the collective dose of workers in a group. For example, using many workers to perform small portions of a task would reduce the individual worker dose to low levels. However, frequent worker changes would make the work inefficient, resulting in a significantly higher collective dose to all the workers than if fewer had received slightly higher individual doses.

SRS worker doses have typically been well below DOE worker exposure limits. DOE set administrative exposure guidelines at a fraction of the exposure limits to help enforce doses that are as low as reasonably achievable. For example, the current DOE worker exposure limit is 5,000 millirem per year, and the 1997 SRS ALARA administrative control level for the whole body is 500 millirem per year. Every year DOE evaluates the SRS ALARA administrative control levels and adjusts them as needed.

Table 3.7-1 lists maximum and average individual doses and SRS collective doses from 1989 to 1998.

TC

Table 3.7-1. SRS annual individual and collective radiation doses.^a

Year	Number with measurable dose	Average individual worker dose (rem) ^b	Site worker collective dose (person-rem)
1989	12,363	0.070	863
1990	11,659	0.065	753
1991	8,391	0.055	459
1992	6,510	0.054	352
1993	5,202	0.051	264
1994	6,284	0.050	315
1995	4,846	0.053	256
1996	4,736	0.053	252
1997	3,327	0.050	165
1998	3,163	0.052	166

a. Adapted from: DOE (1996b); WSRC (1997, 1998, 1999a).

b. The average dose includes only workers who received a measurable dose during the year.

TC

3.7.4 WORKER NONRADIOLOGICAL HEALTH

Industrial hygiene and occupational health programs at the SRS deal with all aspects of worker health and relationship with the work environment. The objective of an effective occupational health program is to protect employees from hazards in their work environment. To evaluate these hazards, DOE uses routine monitoring to determine employee exposure levels to hazardous chemicals.

Exposure limit values are the basis of most occupational health codes and standards. If an overexposure to a harmful agent does not exist, that agent generally does not create a health problem.

The Occupational Safety and Health Administration (OSHA) has established Permissible Exposure Limits to regulate worker exposure to hazardous chemicals. These limits refer to airborne concentrations of substances and represent conditions under which nearly all workers could receive repeated exposures day after day without adverse health effects.

Table 3.7-2 lists the estimated maximum and average annual concentrations of existing OSHA-regulated workplace pollutants modeled in and around existing SRS facilities. Estimated concentration levels for existing OSHA-

regulated workplace pollutants are less than the OSHA Permissible Exposure Limits for all contaminants, with the exception of nitrogen dioxide (as nitrogen oxide) and nitric acid. The large nitrogen dioxide exceedance (a 15-minute average of 406 mg/m³ compared to the OSHA Permissible Exposure Limit of 9 mg/m³) is based on modeling assumptions with maximum potential emissions for diesel units including back-up units operating at ground-level for limited periods (Stewart 1997). The nitric acid value also is based on maximum potential emissions related to conventional processing activities. Actual emissions are expected to be below regulatory limits.

DOE has established industrial hygiene and occupational health programs for the processes covered by this EIS and across the SRS to protect the health of workers from nonradiological hazards.

3.8 Waste and Materials

3.8.1 WASTE MANAGEMENT

This section describes the waste generation baseline that DOE uses in Chapter 4 to gauge the relative impact of each SNF management alternative on the overall production of waste at SRS and on DOE's capability to manage such waste.

Table 3.7-2. Estimated maximum annual concentrations (milligrams per cubic meter) of workplace pollutants regulated by Occupational Safety and Health Administration.^a

Pollutant	OSHA PEL ^b (mg/m ³)	Time period	Concentrations (mg/m ³)	
			Maximum 8-hour average	Annual average
Carbon monoxide	55	8 hours	10	0.53
Nitrogen dioxide (as NO _x)	9	Ceiling limit ^c	406 ^d	2.3
Total particulates	15	8 hours	0.95	0.06
Sulfur dioxide (as SO _x)	13	8 hours	0.63	0.05
Hexane	1,800	8 hours	1.5	0.08
Nitric acid	5	8 hours	11	0.34
Sodium hydroxide	2	8 hours	<0.01	<0.01
Xylene	435	8 hours	136	14.5

a. Source: Stewart (1997).

b. OSHA Permissible Exposure Limits (PEL).

c. Ceiling limits are permissible exposure limits that a facility cannot exceed at any time.

d. 15-minute average.

TC

SRS generates six basic classes of waste – low-level radioactive, high-level radioactive, hazardous, mixed (low-level radioactive and hazardous), transuranic and alpha, and sanitary (nonhazardous, nonradioactive) – which this EIS considers because they are possible byproducts of SNF management. The following sections describe the waste classes. Table 3.8-1 lists projected total waste generation volumes for fiscal years 1999 through 2029 (a 30-year time period encompassing most of the time period of the scenarios addressed in this EIS).

Tables 3.8-2 through 3.8-4 provide an overview of the existing and planned facilities that DOE expects to use in the storage, treatment, and disposal of the various waste classes.

3.8.1.1 Low-Level Radioactive Waste

EC

DOE Order 435.1 (Radioactive Waste Management) defines low-level radioactive waste as radioactive waste that cannot be classified as high-level waste, spent nuclear fuel, transuranic waste, or byproduct material, and that does not have any constituents that are regulated under the Resource Conservation and Recovery Act (RCRA).

At present, DOE uses a number of methods for treating and disposing of low-level waste at SRS, depending on the waste form and activity. Approximately 41 percent of this waste is low in activity and can be treated at the Consolidated Incineration Facility. In addition, DOE could volume-reduce these wastes by compaction, supercompaction, smelting, or repackaging (DOE 1995c). After volume reduction, DOE would package the remaining low-activity waste and place it in either shallow land disposal or vault disposal in E Area.

DOE places low-level wastes of intermediate activity and some tritiated low-level wastes in E Area intermediate activity vaults, and will store long-lived low-level waste (e.g., spent deionizer resins) in the long-lived waste storage buildings in E Area, where they will remain until DOE determines their final disposition.

3.8.1.2 Low-Level Mixed Waste

DOE Order 435.1 defines low-level mixed waste as low-level radioactive waste that contains material listed as hazardous under RCRA or that exhibits one or more of the following hazardous waste characteristics: ignitability, corrosivity, reactivity, or toxicity. It includes such materials

TC

Table 3.8-1. Total waste generation forecast for SRS (cubic meters).^{a,b}

Inclusive Dates	Waste Class				
	Low-level	High-level	Hazardous	Mixed low-level	Transuranic and alpha
1999 to 2029	180,299	14,129	6,315	3,720	6,012

Source: Derived from Halverson (1999).

as tritiated mercury, tritiated oil contaminated with mercury, other mercury-contaminated compounds, radioactively contaminated lead shielding, equipment from the tritium facilities in H Area, and filter paper takeup rolls from the M-Area Liquid Effluent Treatment Facility.

As described in the *Approved Site Treatment Plan* (DOE 1996c), storage facilities for low-level mixed waste are in several different SRS areas. These facilities are dedicated to solid, containerized, or bulk liquid waste and all are approved for this storage under RCRA as interim status or permitted facilities or as Clean Water Act-permitted tank systems. Several treatment processes described in the *Approved Site Treatment Plan* (DOE 1996c) exist or are planned for low-level mixed waste. These facilities, which are listed in Table 3.8-3, include the Consolidated Incineration Facility, the M-Area Vendor Treatment Process, and the Hazardous Waste/Mixed Waste Containment Building.

Depending on the nature of the waste remaining after treatment, DOE plans to use either shallow land disposal or RCRA-permitted hazardous waste/mixed waste vaults for disposal.

3.8.1.3 High-Level Waste

High-level radioactive waste is highly radioactive material from the processing of SNF that contains a combination of transuranic waste and fission products in concentrations that require permanent isolation. It includes both liquid waste produced by processing and solid waste derived from that liquid (DOE 1988).

At present, DOE stores high-level waste in carbon steel and reinforced concrete underground tanks in the F- and H-Area tank farms. The high-level waste undergoes volume reduction by evaporation, and the resulting high activity precipitate is incorporated in borosilicate glass at the Defense Waste Processing Facility Vitrification Facility. The remaining low-activity salt solution is treated and disposed of at the Saltstone Manufacturing and Disposal Facility. Both processes are described in the *Final Supplemental Environmental Impact Statement, Defense Waste Processing Facility* (DOE 1994).

DOE has committed to complete closure by 2022 of the 24 HLW tank systems that do not meet the secondary containment requirements in the Federal Facility Agreement (WSRC 1999a). During waste removal, DOE will retrieve as much of the stored HLW as can be removed using the existing waste transfer equipment. The sludge portion of the retrieved waste will be treated in treatment facilities and vitrified at DWPF. The salt portion of the retrieved waste (processed and treated) will be treated at one of the salt disposition facilities being evaluated in the High-Level Waste Salt Disposition Alternatives EIS (DOE 1999b) and either vitrified at DWPF or disposed as grout in Z Area.

3.8.1.4 Sanitary Waste

Sanitary waste is solid waste that is neither hazardous, as defined by RCRA, nor radioactive. It consists of salvageable material and material that is suitable for disposition in a municipal sanitary landfill. Sanitary waste streams include

Table 3.8-2. Planned and existing waste storage facilities.^a

Storage facility	Location	Capacity	Original waste stream ^b				Status	TC
			Low-level	High-level	Transuranic	Alpha ^c		
Long-lived waste storage buildings	E Area	140 m ³ /bldg	X				One exists.	
Containerized mixed waste storage	Buildings 645-2N, 643-29E, 643-43E, 316-M, and Pad 315-4M	4754 m ³					X	DOE plans to construct additional storage buildings, similar to 643-43E, as necessary.
Liquid mixed waste storage	DWPF Organic Waste Tank (S Area) SRTC Mixed Waste Tanks Liquid Waste Solvent Tanks (H Area) Burial Ground Solvent Tanks (E Area)	9531 m ³					X	The Burial Ground Solvent Tanks are currently undergoing closure. The H-Area Liquid Waste Solvent Tanks were constructed as a re-placement.
High-level waste tank farms	F and H Area	(d)		X				50 underground tanks are currently used for storage ^e .
Failed equipment storage vaults	Defense Waste Processing Facility (S Area)	300 m ³		X				Two exist; DOE plans approximately 12 additional vaults.
Glass waste storage buildings	Defense Waste Processing Facility (S Area)	2,286 canisters		X				One exists; a second is planned for construction in 2007.
Hazardous waste storage facility	Building 710-B Building 645-N Building 645-4N Waste Pad 1 (between 645-2N and 645-4N) Waste Pad 2 (between 645-4N and 645-N) Waste Pad 3 (east of 645-N) Building 316-M	2,501 m ³				X		Currently in use. No additional facilities are planned, as existing space is expected to adequately support the short-term storage of hazardous wastes awaiting treatment and disposal.
Transuranic waste storage pads	E Area	117 m ³ (f)			X	X	X	Currently in use. No additional facilities are planned. 19 pads exist; 10 additional pads may be constructed by 2006.

DWPF = Defense Waste Processing Facility.

SRTC = Savannah River Technology Center.

a. Sources: DOE (1994, 1995a, 1995b, 1996a).

b. Sanitary waste is not stored at SRS, thus it is not addressed in this table.

c. Currently, alpha waste is handled and stored as transuranic waste.

d. Currently the High-level Waste Tanks contain approximately 130,600 m³ of high-level waste. This is almost 90 percent of the usable capacity.

e. Twenty-three of these tanks do not meet secondary containment requirements and have been scheduled for waste removal.

f. Transuranic Pad storage capacities depends on the packaging of the waste and the configuration of packages on the pads.

Table 3.8-3. Planned and existing waste treatment processes and facilities.^a

Waste Treatment Facility	Waste Treatment Process	Waste type					Status
		Low-level	High-level	Transuranic	Alpha ^b	Hazardous	
		Mixed low-level					
Consolidated Incineration Facility	Incineration	X				X	Began treating waste summer 1997
Offsite facility	Smelting	X					Currently ongoing
Defense Waste Processing Facility	Vitrification		X				Currently operational
Defense Waste Processing Facility	Stabilization					X	Currently operational
Replacement high-level waste evaporator ^c	Volume Reduction		X				Radioactive operation anticipated in March 2000
M-Area Vendor Treatment Facility	Vitrification					X	Undergoing Closure
Treatment at point of waste stream origin	Macroencapsulation					X	As feasible based on waste and location
Non-Alpha Vitrification Facility	Vitrification	X				X	Plan to begin operations in 2006
INEEL ^d Waste Engineering Development Facility	Amalgamation/ Stabilization					X	Developing shipping/ treatment schedules
Offsite facility	Offsite Treatment and Disposal					X	Currently ongoing
Offsite facility	Decontamination					X	Plan to begin shipment in FY2000
Various onsite and offsite facilities ^e	Recycle/Reuse	X				X	Currently ongoing
Alpha Vitrification Facility	Vitrification				X		Under evaluation as a potential process
Existing DOE facilities	Repackaging/ Treatment			X			Transuranic waste strategies are still being finalized
M-Area Air Stripper	Air Stripping					X	Currently operational
F- and H-Area Effluent Treatment Facility	Effluent Treatment	X					Currently operational
^{a.} Sources: DOE (1994, 1995a, 1995b, 1996a); WSRC (1995a, 1995b, 1996b); and Odum (1995).							
^{b.} Currently, alpha waste is handled as transuranic waste. After it is assayed and separated, most will be treated and disposed of as low-level or mixed low-level waste.							
^{c.} Evaporation precedes treatment at the Defense Waste Processing Facility and is used to maximize high-level waste storage capacity.							
^{d.} Idaho National Engineering and Environmental Laboratory.							
^{e.} Various waste streams have components (e.g., silver, lead, freon, paper) that might be recycled or reused. Some recycling activities might occur onsite, while other waste streams are directed offsite for recycling. Some of the recycled products are released for public sale, while others are reused onsite.							

a. Sources: DOE (1994, 1995a, 1995b, 1996a); WSRC (1995a, 1995b, 1996b); and Odum (1995).

b. Currently, alpha waste is handled as transuranic waste. After it is assayed and separated, most will be treated and disposed of as low-level or mixed low-level waste.

c. Evaporation precedes treatment at the Defense Waste Processing Facility and is used to maximize high-level waste storage capacity.

d. Idaho National Engineering and Environmental Laboratory.

e. Various waste streams have components (e.g., silver, lead, freon, paper) that might be recycled or reused. Some recycling activities might occur onsite, while other waste streams are directed offsite for recycling. Some of the recycled products are released for public sale, while others are reused onsite.

Table 3.8-4. Planned and existing waste disposal facilities.^a

	Disposal facility	Location	Capacity (m ³)	Original waste stream ^b					Status
				Low-level	High-level	Transuranic	Hazardous	Mixed Low-level	
TC	Shallow land disposal trenches	E Area	(c)	X					Four have been filled; up to 58 more may be constructed.
EC	Low-activity vaults	E Area	30,500/vault	X					One vault exists and one additional is planned.
	Intermediate-activity vaults	E Area	5,300/vault	X					Two vaults exist and five more may be constructed.
TC	Hazardous waste/mixed waste vaults	NE of F Area	2,300/vault				X	X	RCRA permit application submitted for 10 vaults. At least 11 additional vaults may be needed.
	Saltstone Disposal Facility	Z Area	80,000/vault ^d	X					Two vaults exist and ap- proximately 13 more are planned.
EC	Three Rivers Landfill	SRS Intersection of SC 125 and Rd. 2	NA						Current destination for SRS sanitary waste.
	Burma Road Cellulosic and Construction Waste Landfill	SRS Intersection of C Rd. and Burma Rd	NA						Current destination for demolition/construction debris. DOE expects to reach permit capacity in 2008.
TC	Waste Isolation Pilot Plant (WIPP)	New Mexico	175,600			X			EPA certification of WIPP completed in April 1998. RCRA certification finalized in 1999. ^e
EC	Federal repository	See Status	NA		X				Proposed Yucca Mountain, Nevada site is currently under investigation.

RCRA = Resource Conservation and Recovery Act.

NA = Not Available.

a. Sources: DOE (1994, 1995a, 1995b, 1995c, 1996a, 1996c); WSRC (1995a and 1996b).

b. After alpha waste is assayed and separated from the transuranic waste, DOE plans to dispose of it as low-level or mixed low-level waste so it is not addressed separately here.

c. Various types of trenches exist including engineered low-level trenches, greater confinement disposal boreholes and engineered trenches, and slit trenches. The different trenches are designed for different waste types, are constructed differently, and have different capacities.

d. This is the approximate capacity of a double vault. One single vault and one double vault have been constructed. Future vaults are currently planned as double vaults.

e. SRS is scheduled for WIPP certification audit in 2000, after which WIPP could begin receiving SRS waste.

such items as paper, glass, discarded office material, and construction debris (DOE 1994).

Sanitary waste volumes have declined due to recycling and the decreasing SRS workforce. DOE sends sanitary waste that is not recycled or reused to the Three Rivers Landfill on SRS. The SRS also continues to operate the Burma Road Cellulosic and Construction Waste Landfill to dispose of demolition and construction debris.

3.8.1.5 Hazardous Waste

Hazardous waste is nonradioactive waste that SCDHEC regulates under RCRA and corresponding state regulations. Waste is hazardous if the EPA lists it as such or if it exhibits the characteristic(s) of ignitability, corrosivity, reactivity, or toxicity. SRS hazardous waste streams consist of a variety of materials, including mercury, chromate, lead, paint solvents, and various laboratory chemicals.

At present, DOE stores hazardous wastes in three buildings and on three solid waste storage pads that have RCRA permits. Hazardous waste is sent to offsite treatment and disposal facilities, and could be treated at the Consolidated Incineration Facility in the future. DOE also plans to continue to recycle, reuse, or recover certain hazardous wastes, including metals, excess chemicals, solvents, and chlorofluorocarbons. Wastes remaining after treatment might be suitable for either shallow land disposal or disposal in the Hazardous/Mixed Waste Disposal Vaults (DOE 1995c).

3.8.1.6 Transuranic and Alpha Waste

Transuranic waste contains alpha-emitting transuranic radionuclides (those with atomic weights greater than 92) that have half-lives greater than 20 years at activities exceeding 100 nanocuries per gram (DOE 1988). At present, DOE manages low-level alpha-emitting waste with activities between 10 and 100 nanocuries

per gram, referred to as alpha waste, as transuranic waste at SRS.

The SRS Waste Management EIS (DOE 1995c) describes the handling and storage of transuranic and alpha waste at the SRS. This consists primarily of providing continued safe storage until treatment and disposal facilities are available.

The *Strategic Plan for Savannah River Site Transuranic Waste* (WSRC 1996b) defines the future handling, treatment and disposal of the SRS transuranic and alpha waste stream. Eventually, DOE plans to ship the transuranic and mixed transuranic waste to the Waste Isolation Pilot Plant in New Mexico for disposal.

Before disposition, DOE plans to assay the wastes stored on the pads and segregate the alpha waste. Vitrification is an option for at least part of the mixed alpha waste (DOE 1996b). Following assay, DOE could dispose of much of the alpha waste as either mixed low-level or low-level waste.

3.8.2 HAZARDOUS MATERIALS

The *Savannah River Site Tier Two Emergency and Hazardous Chemical Inventory Report* for 1998 (WSRC 1999b) lists more than 79 hazardous chemicals that were present at SRS at some time during the year in amounts that exceeded the minimum reporting thresholds [10,000 pounds (4,536 kilograms) for hazardous chemicals and 500 pounds (227 kilograms) or less for extremely hazardous substances]. Four of the 79 are extremely hazardous substances under the Emergency Planning and Community Right-to-Know Act of 1986. The actual number and quantity of hazardous chemicals present on the Site and at individual facilities changes daily as a function of use and demand.

TC

TC

References

- Aadland, R. K., J. A. Gellici, and P. A. Thayer, 1995, *Hydrogeologic Framework of West-Central South Carolina*, Report 5, State of South Carolina Department of Natural Resources - Water Resources Division, Columbia, South Carolina.
- Aiken Standard, 1993, "Early Sunday Quake Jolts Aiken County," Volume 26, No. 221, Aiken, South Carolina, August 9. TC
- Arnett, M. W., L. K. Karapatakis, and A. R. Mamatey, 1993, *Savannah River Site Environmental Report for 1992*, WSRC-TR-93-075, Westinghouse Savannah River Company, Environmental Protection Department, Environmental Monitoring Section, Aiken, South Carolina.
- Arnett, M. W., L. K. Karapatakis, and A. R. Mamatey, 1994, *Savannah River Site Environmental Report for 1993*, WSRC-TR-94-075, Westinghouse Savannah River Company, Aiken, South Carolina.
- Arnett, M. W., L. K. Karapatakis, A. R. Mamatey, and J. L. Todd, 1992, *Savannah River Site Environmental Report for 1991*, WSRC-TR-92-186, Westinghouse Savannah River Company, Aiken, South Carolina.
- Arnett, M. W., and A. R. Mamatey, 1996, *Savannah River Site Environmental Report for 1995*, WSRC-TR-96-0075, Westinghouse Savannah River Company, Aiken, South Carolina.
- Arnett, M. W., and A. R. Mamatey, 1997, *Savannah River Site Environmental Report for 1996*, WSRC-TR-97-0170, Westinghouse Savannah River Company, Aiken, South Carolina.
- Arnett, M. W., and A. R. Mamatey, 1998a, *Savannah River Site Environmental Data for 1997*, WSRC-TR-97-00324, Westinghouse Savannah River Company, Aiken, South Carolina. TC
- Arnett, M. W., and A. R. Mamatey, 1998b, *Savannah River Site Environmental Report for 1997*, WSRC-TR-97-00322, Westinghouse Savannah River Company, Aiken, South Carolina.
- Bollinger, G. A., 1973, "Seismicity of the Southeastern United States," *Bulletin of the Seismological Society of America*, 63, 5, pp. 1785-1808.
- CDC (Centers For Disease Control and Prevention), 1998, *National Vital Statistics Report*, Volume 47, No. 4, U.S. Public Health Service, Department of Health and Human Services, Washington, D.C., October 7. TC
- CEQ (Council on Environmental Quality), 1998, *Environmental Justice Guidance Under the National Environmental Policy Act*, Washington, D.C.
- Cothran, E. G., M. H. Smith, J. O. Wolff, and J. B. Gentry, 1991, *Mammals of the Savannah River Site*, SRO-NERP-21, Savannah River Ecology Laboratory, Aiken, South Carolina.
- Cummins, C. L., D. K. Martin, and J. L. Todd, 1990, *Savannah River Site Environmental Report for 1989*, WSRC-IM-90-60, Westinghouse Savannah River Company, Aiken, South Carolina.

Cummins, C. L., D. K. Martin, and J. L. Todd, 1991, *Savannah River Site Environmental Report for 1990*, Volumes I-II, WSRC-IM-91-28, Westinghouse Savannah River Company, Aiken, South Carolina.

DOE (U.S. Department of Energy), 1987, *Final Environmental Impact Statement, Waste Management Activities for Groundwater Protection*, Savannah River Plant, Aiken, South Carolina, DOE/EIS-0120, Savannah River Operations Office, Aiken, South Carolina.

DOE (U.S. Department of Energy), 1988, *Radioactive Waste Management*, DOE Order 5820.2A, Washington D.C.

DOE (U.S. Department of Energy), 1990a, *Final Environmental Impact Statement, Continued Operation of K-, L-, and P-Reactors, Savannah River Site, Aiken, South Carolina*, DOE/EIS-0147, Volume I, Savannah River Operations Office, Aiken South Carolina.

DOE (U.S. Department of Energy), 1990b, Order 5400.5, "Radiation Protection of the Public and the Environment," Washington, D.C., February 8.

DOE (U.S. Department of Energy), 1994, *Final Defense Waste Processing Facility Supplemental Environmental Impact Statement*, DOE/EIS-0082-S, Savannah River Operations Office, Aiken, South Carolina.

DOE (U.S. Department of Energy), 1995a, *Final Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement*, DOE/EIS-0203, Idaho Operations Office, Idaho Falls, Idaho.

DOE (U.S. Department of Energy), 1995b, *Final Environmental Impact Statement, Interim Management of Nuclear Materials*, DOE/EIS-0220, Savannah River Operations Office, Aiken, South Carolina.

DOE (U.S. Department of Energy), 1995c, *Savannah River Site Waste Management Final Environmental Impact Statement*, DOE/EIS-0217, Savannah River Operations Office, Aiken, South Carolina.

DOE (U.S. Department of Energy), 1996a, *Final Environmental Impact Statement: Shutdown of the River Water System at the Savannah River Site*, DOE/EIS-0268, Savannah River Operations Office, Aiken, South Carolina.

DOE (U.S. Department of Energy), 1996b, *The United States Department of Energy's Web Page for Information on Occupational Radiation Exposure: DOE Radiation Exposure Monitoring Systems* [web page; updated 10/25/96], <http://rems.eh.doe.gov/rems.htm>. [accessed 3/25/98].

TC | DOE (U.S. Department of Energy), 1996c, *Approved Site Treatment Plan*, Revision 4, Savannah River Operations Office, Aiken, South Carolina.

TC | DOE (U.S. Department of Energy), 1998, *Final Environmental Impact Statement for Construction and Operation of a Tritium Extraction Facility at the Savannah River Site*, DOE/EIS-0271, Savannah River Operations Office, Aiken, South Carolina, May.

TC | DOE (U.S. Department of Energy), 1999a, *High-Level Waste Tank Closure Draft Environmental Impact Statement*, DOE/EIS-0303D, Savannah River Operations Office, Aiken, South Carolina.

- DOE (U.S. Department of Energy), 1999b, *High-Level Waste Salt Disposition Alternatives Environmental Impact Statement*, DOE/EIS-0082-S2D, Savannah River Operations Office, Aiken, South Carolina. | TC
- EPA (U.S. Environmental Protection Agency), 1993, "Office of Environmental Equity Grants Program; Solicitation Notice for Fiscal Year (FY) 1994, Environmental Justice Grants to Community Groups," 58 FR 63955, December 3.
- EPA (U.S. Environmental Protection Agency), 1994, *Environmental Justice Initiatives 1993*" EPA 200-R-93-001, Washington, D.C.
- EPA (U.S. Environmental Protection Agency), 1998, *Guidance for Incorporating Environmental Justice Concerns in EPA's NEPA Compliance Analysis*, Office of Federal Activities, Washington, D.C.
- EPA (U.S. Environmental Protection Agency), 1999, June 28, 1999 EPA Summary of the Court's Decision, <http://ttnwww.rt-prc.epa.gov/naaqsfinl>, July 19. | TC
- Gibbons, J. W. and R. D. Semlitsch, 1991, *Guide to the Reptiles and Amphibians of the Savannah River Site*, University of Georgia Press, Athens, Georgia.
- Halverson, N. V., L. D. Wike, K. K. Patterson, J. A. Bowers, A. L. Bryan, K. F. Chen, C. L. Cummins, B. R. del Carmen, K. L. Dixon, D. L. Dunn, G. P. Friday, J. E. Irwin, R. K. Kolka, H. E. Mackey, Jr., J. J. Mayer, E. A. Nelson, M. H. Paller, V. A. Rogers, W. L. Specht, H. M. Westbury, and E. W. Wilde, 1997, *SRS Ecology Environmental Information Document*, WSRC-TR-97-0223, Westinghouse Savannah River Company, Aiken, South Carolina.
- Halverson, N., 1999, interoffice memorandum to B. Shedrow, "Revised Cumulative Impacts Data," SRT-EST-99-0328, Westinghouse Savannah River Company, Aiken, South Carolina. | TC
- HNUS (Halliburton NUS Corporation), 1997, *Socioeconomic Characteristics of Selected Counties and Communities Adjacent to the Savannah River Site*, Aiken, South Carolina.
- Hunter, C. H., 1990, *A Climatological Description of the Savannah River Site*, WSRC-RP-89-313, Westinghouse Savannah River Company, Aiken, South Carolina. | TC
- NCRP (National Council on Radiation Protection and Measurements), 1987a, *Ionizing Radiation Exposure of the Population of the United States*, Report No. 93, Washington, D.C.
- NCRP (National Council on Radiation Protection and Measurements), 1987b, *Exposure of the Population in the United States and Canada from Natural Background Radiation*, Report No. 94, Washington, D.C.
- NCRP (National Council on Radiation Protection and Measurements), 1993, *Limitation of Exposure to Ionizing Radiation*, Report No. 116, Washington, D.C.
- NOAA (National Oceanic and Atmospheric Administration), 1994, *Local Climatological Data 1993 Annual Summary for Augusta, Georgia*, National Climatic Data Center, Asheville, North Carolina.
- NRC (U.S. Nuclear Regulatory Commission), 1996, *Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites in 1992*, NUREG/CR-2850, Vol. 14, Washington, D.C.

- Odum, J. V., 1995, "Storage of Wastes Subject to the Land Disposal Restrictions at the Savannah River Site Pending Radiological Analysis," letter to D. Wilson (Bureau of Solid and Hazardous Waste Management, South Carolina Department of Health and Environmental Control), Westinghouse Savannah River Company, Aiken, South Carolina, December 21.
- Sassaman, K., 1997a, "Archaeological Review of Proposed Spent Fuel Facilities at C and L Reactors," memorandum to L. S. Moore (Halliburton NUS Corporation), Savannah River Archaeological Research Program, South Carolina Institute of Archaeology and Anthropology, Aiken, South Carolina.
- Sassaman, K., 1997b, "Archaeological Review of Proposed Spent Fuel Facilities at P-Reactor," memorandum to L. S. Moore (Halliburton NUS Corporation), Savannah River Archaeological Research Program, South Carolina Institute of Archaeology and Anthropology, Aiken, South Carolina.
- SCDHEC (South Carolina Department of Health and Environmental Control), 1976, "Air Pollution Control Regulations and Standards," Regulation 61-62.5, pursuant to Section 48-1-30 through 48-1-60 of the 1976 South Carolina Code of Laws, as amended, Columbia, South Carolina.
- SCDHEC (South Carolina Department of Health and Environmental Control), 1992, *South Carolina Nuclear Facility Monitoring - Annual Report 1992*, Columbia, South Carolina.
- Shedrow, B., 1993, *SRS Affected Environment*, WSRC-TR-93-667, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Sprunt, A. S., and E. B. Chamberlain, 1970, *South Carolina Bird Life*, Second Edition, University of South Carolina Press, Columbia, South Carolina.
- SRFS (Savannah River Forest Station), 1994, *Savannah River Site Proposed, Threatened, Endangered, and Sensitive Plants and Animals*, Forest Service, U.S. Department of Agriculture, Aiken, South Carolina.
- SRFS (Savannah River Forest Station), 1996, *Savannah River Site Threatened, Endangered, and Sensitive Plants and Animals* (map), Savannah River Site, Aiken, South Carolina.
- SRFS (Savannah River Forest Station), 1997, *Stand Prescription Summaries for Timber Compartments 71 and 74*, Savannah River Site, Aiken, South Carolina.
- Stephenson, D. E. and A. L. Stieve, 1992, Structured Model of the Basement in the Central Savannah River Area, South Carolina and Georgia, WSRC-TR-92-120, Westinghouse Savannah River Company, Aiken, South Carolina.
- Stewart, J., 1997, "HNUS Data Request for SNF EIS," SRT-NTS-97-0099, interoffice memorandum to C. B. Shedrow, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina, June 6.
- URS/Blume (URS/John Blume and Associates, Engineers), 1982, *Update of Seismic Criteria for the Savannah River Plant, Volume I: Geotechnical*, DPE-3699, prepared for E. I. du Pont de Nemours and Company, Inc., Savannah River Plant, Aiken, South Carolina.

U.S. Bureau of the Census, 1990a, *Summary Tape File 1B*, Table P8, Washington, D.C.

U.S. Bureau of the Census, 1990b, *Summary Tape File 1B*, Table 117, Washington, D.C.

U.S. Bureau of the Census, 1999, *County Population Estimates for July 1, 1982* [on line], <http://www.census.gov/population/estimates/county>, release date March 12.

TC

Weber, et al., 1998; *Tornado Maximum Wind Gust and Extreme Rainfall Event Recurrence Frequencies at the Savannah River Site (U)*, WSRC-TR-98-00329, Westinghouse Savannah River Company, Aiken, South Carolina.

Wells, D., 1997, personal communication with G. Gunter (Halliburton NUS Corporation), Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina, February 6.

Wike, L. D., D. B. Moore-Shedrow, and C. B. Shedrow, 1996, *Site Selection for the Accelerator for Production of Tritium at the Savannah River Site*, WSRC-TR-96-0279 Rev. 1, Westinghouse Savannah River Company, Aiken, South Carolina, October 9.

WSRC (Westinghouse Savannah River Company), 1993, *Nuclear Weapons Complex Reconfiguration PEIS Data Report for the Savannah River Site, No-Action Alternative*, ESH-NEP-93-0188, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1994, *Air Dispersion Modeling for the Spent Nuclear Fuel Environmental Impact Statement - Nonradiological Emission*, WSRC-RP-94-147 (Draft), Savannah River Site, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1995a, *The Savannah River Site's Groundwater Monitoring Program*, First Quarter through Fourth Quarter 1995, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1995b, *High-Level Waste System Plan*, Revision 6, LW-OVP-95-0102, High-Level Waste Management Division, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1996a, *Preliminary Site Selection for the Spent Nuclear Fuel Transfer and Storage Facility*, WSRC-TR-96-0398 (Draft), Savannah River Site, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1996b, *Strategic Plan for Savannah River Site (SRS) Transuranic (TRU) Waste*, Revision 0, WSRC-RP-96-482, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1997, *Radiological Performance (U)*, Fourth Quarter 1996, WSRC-SHP-970012, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1998, *Savannah River Site Radiological Performance, 4th Quarter 1997*, ESH-SHP-980007, Savannah River Site, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1999a, *Radiological Performance Indicators (U)*, Fourth Quarter CY98, ESH-HPT-99-0017, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1999b, *Savannah River Site Tier II Emergency and Hazardous Chemical Inventory Report*, Aiken, South Carolina.

TC

CHAPTER 4. ENVIRONMENTAL IMPACTS

This chapter describes potential environmental impacts from construction, operation, and accidents associated with the proposed action and its alternatives. Section 4.1 describes the operational impacts of each alternative within the scope of this environmental impact statement (EIS). Section 4.2 describes risks to members of the public and onsite workers from potential facility accidents associated with the management of spent nuclear fuel (SNF) at the Savannah River Site (SRS). Section 4.3 describes impacts that could result from construction activities associated with SNF management at SRS. The purpose of the information presented in this chapter is to provide comparisons among alternatives. For new facilities, this information is based on DOE's best estimates of these facilities' operational characteristics. These data are not intended to be used for safety analysis purposes or compared to safety documents such as a Safety Analysis Report.

As discussed in Section 2.3.2, the Department of Energy (DOE) has identified three candidate sites for the potential construction of a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility: (1) the east side of L Area inside the facility fence, (2) the southeast side of C Area inside the facility fence, and (3) the northeast side of P Area. In addition, the facility could be constructed on a site inside the F-Area or H-Area fence or in an existing reactor building such as Building 105-L.

In most instances, implementing the technology options described in Chapter 2 would result in the same or very similar environmental impacts, regardless of location. If, during the preparation of this EIS, analyses indicated that a technology option would produce different environmental impacts at one of the candidate sites, DOE analyzed the site that would have the greatest impact (the bounding site). The analysis of the atmospheric releases of radioactivity described in the air resources and public and worker health sections is based on the assumption that emissions from a Transfer and Storage Facility or

Transfer, Storage, and Treatment Facility would occur in C Area. Releases from C Area would result in higher estimated radiation doses to members of the public than releases from L or P Area (i.e., C Area would result in doses to the maximally exposed offsite individual approximately 1.7 times higher than those in L Area and 1.1 times higher than those in P Area). All other impacts would be independent of location.

The impacts reported in this chapter are based on the entire SNF inventory described in Chapter 1 and Appendix C. However, as noted in Section 1.3, some foreign reactor operators may not participate in DOE's program of accepting U.S.-origin SNF. This reduction in receipts could potentially impact the amounts of fuel in Groups B, D, and E. Therefore, the amounts of fuel to be managed in those fuel groups could be less than the amounts assumed for the calculations in Chapter 4. DOE believes that annual impacts for normal operations, construction impacts, and accident impacts would be unaffected by modest reductions in the expected fuel inventory. The annual impacts are based on the maximum year's impacts; decreasing the foreign fuel shipments may lessen the number of years of fuel handling, conditioning, or treatment, but would not affect the maximum annual impact. SNF accidents usually involve small amounts of fuel and thus are insensitive to the total inventory. Construction impacts are similarly insensitive to the reduction in total fuel inventory that could occur. Eleven environmental impact measures are based on activities that occur over the entire period of analysis. These impacts would be sensitive to reductions in fuel receipts. Where applicable, the tables in this chapter explain how to adjust reported impacts for potentially reduced fuel receipts.

4.1 Impacts from Normal Operations

This section describes environmental impacts that could result from operational activities as-

sociated with SNF management at SRS for existing and new facilities. Because the only potential impacts to geologic and cultural resources would occur during construction (see Section 4.3), Section 4.1 does not consider geologic or cultural resource impacts. DOE does not anticipate a significant increase in employment due to the implementation of any technology options (Table 4.1-1). The existing site work force should be sufficient to provide the necessary operations and support personnel; therefore, there would be no socioeconomic impacts from operations under any technology.

Table 4.1-1. Estimated operational staffing for any of the technology options.

Technology option	Operations personnel	Support personnel	Total personnel
Melt and Dilute	200	200	400
Mechanical Dilution	175	175	350
Repackage and Prepare to Ship	75	75	150
Vitrification	317	317	634
Electrometallurgical	238	238	476
Conventional Processing	300	300	600
Continued Wet Storage	80	80	160

Source: Bickford et al. 1997.

DOE used the following process to estimate the impacts associated with new facilities/processes. First, DOE identified the facilities that would be needed to implement each of the technologies described in Chapter 2 (see Table 2-4). Next, DOE identified the major systems required within each facility for each technology. DOE then identified the energy sources, potential waste and effluent streams, and sources of potential radiation exposure associated with each of these major systems. These results were then compared to similar processes with which DOE has operational experience to determine the relative magnitude of the impact. These impacts were presented as annual impacts; integrated

impacts were then calculated as described below in Section 4.1.1.

DOE does not expect normal operations to have any appreciable impacts on ecological resources. Impacts would be limited to minor disturbances of animals in undeveloped areas adjacent to SNF management facilities caused by increased movement and noise from personnel, vehicles, and equipment. However, these impacts would be negligible under all proposed technology options because they would occur in areas where industrial activities already exist. Impacts to potential human receptors from normal releases of radioactive and nonradioactive contaminants to the environment would be small for any of the technologies under consideration (Section 4.1.1.3). Therefore, these releases would not be likely to produce measurable effects on nearby plant and animal communities or to accumulate in aquatic or terrestrial ecosystems.

4.1.1 IMPACTS OF TECHNOLOGY OPTIONS

This section describes the environmental impacts of each technology. The analysis covers the environmental impacts of actions over the 38-year period from 1998 through 2035 and presents both maximum annual impacts from these technologies and estimated total impacts over the entire period. For example, the discussions of water and air resources present maximum annual radiation doses to members of the public from liquid and airborne emissions associated with each technology and compares the resulting values to Federal limits. The section on public and worker health, on the other hand, presents radiation doses to members of the public from liquid and airborne emissions over the entire implementation period. The waste generation and utilities and energy sections also present impacts over the entire period of analysis (1998-2035).

To estimate total impacts, DOE identified the activities necessary to implement each technology, the amount of time required for each step (*phase*) of the technology option, and the annual

impacts likely to occur during each phase. DOE summed the annual impacts over the entire duration of the phase, together with other phases needed to implement that option. For the Conventional Processing option, DOE used historic data for F- and H-Canyon operations to estimate the time needed to process the entire inventory of each type of fuel (McWhorter 1997). For the other technology options with a treatment phase, DOE used engineering judgments to estimate the duration of this phase for each fuel group. Appendix E describes the assumed durations for each phase. If annual impact data (i.e., utilities and energy, waste generation, and worker radiation dose) for each type of fuel were not available, DOE assumed that the fraction of the impact attributable to each type of fuel would be equal to the fraction of that fuel's fissile mass to the total fissile mass of SNF in the scope of this EIS. DOE derived the annual impact calculations from the available data (Bickford et al. 1997) based on the total radionuclide inventory for each type of fuel. Appendix C contains the radionuclide inventories, using a "reference fuel assembly" i.e., a conservative estimate of the radionuclide and curie content for an SNF assembly designed to bound the characteristics of fuel assigned to SRS. The engineering report that provides data upon which the impacts presented in this chapter are based (Bickford et al. 1997) is available for review at the DOE public reading room in Aiken, South Carolina.

4.1.1.1 Water Resources

This section describes the effects of normal operations associated with the technologies to SRS waters. All process water would come from groundwater. None of the technologies require much water to process the fuels. At most, less than 6,000 liters per year (equivalent to 1,585 gallons per year) would be required. The SRS annually withdraws more than 5×10^9 liters of groundwater (DOE 1997).

As discussed below, the only technology that would result in discharges of radionuclides or nonradioactive hazardous materials to surface water would be conventional processing. The major sources of liquid effluents from facilities

associated with conventional processing would be process cooling water and steam condensate that could contain small quantities of radionuclides and chemicals. Conventional processing would use wastewater treatment facilities and other equipment designed for full production (i.e., five production reactors, two separation facilities, and other industrial facilities) loads. Therefore, capacities would be sufficient to handle the liquid effluents and other secondary waste associated with conventional processing.

Liquid effluents associated with the SNF technologies would use existing wastewater treatment facilities and outfalls described in Section 3.2.1.3. Sanitary waste would be treated at the SRS Central Wastewater Treatment Facility (CSWTF) and discharged through an existing NPDES outfall (G-10). Because technology options would not increase the number of permanent SRS employees, the CSWTF treatment rates would not be affected, and it would continue to meet the requirements of the SRS NPDES permit.

DOE evaluated in the Programmatic SNF EIS (DOE 1995b) the potential impacts to groundwater from a direct leak to the subsurface from a breach in a storage pool during routine operations. Because basin water could contain some radionuclides but would not contain any toxic or harmful chemicals, the following evaluation addresses only the consequences of radionuclide releases. The analysis conservatively assumed a 5-gallon (19-liter) per-day leak as a result of secondary containment or piping failure at the Receiving Basin for Offsite Fuels, L-Reactor Disassembly Basin, or a new wet receipt basin in a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility. The analysis assumed further that the leak would go undetected for 1 month.

The reliability and sensitivity of the leak detection devices at a new wet receipt basin would be equal to or superior to those required by the U.S. Nuclear Regulatory Commission (NRC 1975) for SNF storage facilities in commercial nuclear power plants. Constant process monitoring, mass balance, and facility design (in-

cluding double-walled containment of vessels and piping) also would be used by DOE to limit operational releases from a new wet receipt facility to near zero.

A leak from the Receiving Basin for Offsite Fuels, or the L-Reactor Disassembly Basin, could result in the introduction of radionuclide-contaminated water into the ground at depths as much as 44 feet (13.4 meters) below grade. Such a release would go directly to the uppermost aquifer (Upper Three Runs), which at SRS is not suitable for use as a drinking water source because of its low yield and the presence of contaminants. Any contaminants would move through the Upper Three Runs and Gordon aquifers and ultimately discharge to SRS streams. The processes governing the plume movement (i.e., the hydraulic conductivity, hydraulic gradient, and effective porosity of aquifers in F, H, and the Reactor Areas) and the processes resulting in the attenuation of contaminants and radionuclides (i.e., radioactive decay, trapping of particulates in the soil, ion exchange in the soil, and adsorption to soil particles) would mitigate impacts to surface- or groundwater resources. Localized contamination of groundwater in the surface aquifer could occur in the immediate vicinity of the storage facility. However, this aquifer is not used as a source of drinking water. DOE concludes that no radionuclide contamination of deeper confined aquifers that are sources of onsite or offsite drinking water would be likely to occur from a leak in a storage basin.

The aquifer used as the primary source for drinking water is separated from the shallower aquifers by a confining unit. The hydraulic pressure of the lower aquifer is greater than that of the overlying aquifer. Therefore, water flows from the lower to the upper aquifer. This upward flow would prevent the downward migration of released contaminants.

4.1.1.1.1 Radiological Impacts

With the exception of conventional processing which is the maximum impact alternative, none of the technologies proposed in this EIS is likely

to result in measurable increases in radionuclides released to water (Bickford et al. 1997). No other proposed technology would have a process discharge to surface waters.

The prolonged storage of SNF in the basins (i.e., the No-Action Alternative) could lead to a higher rate of fuel failures and releases to basin water, but probably would not affect routine releases (i.e., those from national pollutant discharge elimination system [NPDES] permitted outfalls). DOE would maintain water quality by monitoring basin water, deionizing basin water using resin beds, and stabilizing leaking assemblies.

Calculations of radiological doses through water pathways based on these releases are supported by the use of LADTAPXL, a spreadsheet version of the LADTAP II computer code developed by the U.S. Nuclear Regulatory Commission (NRC) to estimate radiation doses associated with normal reactor system liquid effluent releases to individuals, populations, and biota (Hamby 1991). LADTAP II uses the models in NRC Regulatory Guide 1.109 (NRC 1977) to calculate doses received from water and fish ingestion and from recreational water activities. Parameters used to calculate dose for the maximally exposed individual are consistent with regularly published SRS environmental reports (e.g., Arnett and Mamatey 1996).

Any radionuclide releases to surface water resulting from the technologies would be to SRS streams that discharge to the Savannah River. For all technology options, the ingestion of fish contaminated with cesium-137 would contribute most of the exposure to both the maximally exposed individual and the population. Plutonium and uranium isotopes ingested with drinking water would be smaller contributors for the approximately 70,000 people served by water treatment plants near Port Wentworth, Georgia (60,000) and Beaufort, South Carolina (10,000) (Arnett and Mamatey 1996). Table 4.1-2 lists both the maximally exposed individual dose and the collective dose due to liquid releases to the 620,100-person population surrounding SRS.

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Table 4.1-2. Estimated maximum incremental annual dose to hypothetical maximally exposed individual and 620,100-person population surrounding SRS due to liquid releases from Conventional Processing.

Fuel group	MEI dose (millirem)	Population dose (person-rem)
A. Uranium and Thorium Metal Fuels	4.2×10^{-5}	2.4×10^{-4}
B. Materials Test Reactor-Like Fuels	0.042	0.14
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	0.014	0.047
D. Loose Uranium Oxide in Cans	1.4×10^{-3}	4.7×10^{-3}
E. Higher Actinide Targets	NA	NA
F. Non-Aluminum-Clad Fuels	NA	NA

NA = Technology is not applicable to this fuel type.

HEU = Highly Enriched Uranium.

LEU = Low Enriched Uranium.

MEI = Maximally Exposed Individual.

4.1.1.1.2 Nonradiological Impacts

This assessment compared chemical releases with applicable water quality standards. These standards are based on the preservation of aquatic biota populations, human health, and aesthetics (i.e., taste and odor). Figure 3.2-1 shows that conventional processing activities would not occur in the 100-year floodplain. DOE would treat sanitary waste generated by any of the alternatives in this EIS in existing sewage treatment facilities; discharges from these facilities would continue to meet NPDES permit limits.

Activities associated with the New Packaging Technology options and all new treatment options under the New Processing Technology, including Melt and Dilute, Mechanical Dilution, Vitrification, and Electrometallurgical Treatment, would conform to current regulatory standards, and would not have nonradiological waterborne releases (Bickford et al. 1997). Under conventional processing, process cooling water treatment would result in releases of the following concentrations from F Area to Upper Three Runs:

- Nitrate - 40 micrograms per liter
- Ammonia - 30 micrograms per liter
- Manganese - 10 micrograms per liter
- Uranium - 20 micrograms per liter

- Nickel - 50 micrograms per liter
- Chromium - 20 micrograms per liter
- Aluminum - 200 micrograms per liter
- Copper - 10 micrograms per liter
- Zinc - 70 micrograms per liter

Similar or lower concentrations would be released from H Area with the exception of those for nitrate and ammonia, which would be 100 and 500 micrograms per liter, respectively.

Although proposed or final Federal drinking water standards do not apply to discharges, the SRS discharge concentrations would not exceed these standards. The discharges would also comply with South Carolina Water Quality Standards contained in South Carolina Regulation R.61-68. In general, the release concentrations would be no greater than those currently measured in Upper Three Runs and Fourmile Branch (Arnett 1996), with the exception of zinc and ammonia; however, zinc concentrations in the discharge would be only a small fraction of the South Carolina Water Quality Standards, which are based on the taste and odor of drinking water. Ammonia concentrations in the discharge (only H-Area releases would increase current stream concentrations) would be well within state standards. Lead, nickel, and chromium generally were not detected in Upper Three Runs and Fourmile Branch in 1995.

4.1.1.2 Air Resources

This section describes incremental air quality impacts from nonradiological and radiological emissions for the operation of each technology option for each fuel group; this description includes impacts to on- and offsite individuals and populations.

This analysis presents results in terms of ground-level air concentrations for nonradiological constituents and radiation dose for radionuclides because these are the best measures of potential adverse human health effects.

4.1.1.2.1 Nonradiological Emissions

DOE estimated nonradiological emission rates for each technology option (Bickford et al. 1997) and used them with the meteorological data described in Section 3.3.1 to estimate site boundary and noninvolved worker concentrations. This analysis assumed average meteorological conditions.

Onsite Concentrations

The purpose of this analysis is to estimate air concentrations to which SRS workers not involved in SNF management and related operations would be exposed. Atmospheric emissions would occur from F or H Area (conventional processing), L-Reactor Disassembly Basin and the Receiving Basin for Offsite Fuels (continued wet storage), and the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility. To determine impacts to noninvolved workers, the analysis used a generic location 2,100 feet (640 meters) from the release in the direction of the plume of greatest concentration. The 2,100-foot criterion is based on NRC guidance. Also, the use of this distance ensures consistency between this and previous SRS EISs.

The analysis assumed that operational nonradiological releases would be from the same release stack as radiological releases. In addition, this EIS does not include onsite concentrations at distances greater than 2,100 feet; the analysis

considered such concentrations and found that they would be less than those at 2,100 feet.

Tables F-1 through F-10 in Appendix F list estimated air concentrations above baseline (i.e., incremental increases) resulting from nonradiological atmospheric emissions associated with SNF fuel groups. No incremental atmospheric emissions above the baseline presented in Chapter 3 would be associated with Repackage and Prepare to Ship, the only option applicable to the non-aluminum-clad fuels. The air quality regulatory standards listed in Tables F-6 through F-10 in Appendix F are applicable to the Site boundary concentration from all SRS emissions. While these standards are included only for reference, all the incremental concentrations from SNF activities would be at least two orders of magnitude less than any of the corresponding standards except those for nitric acid, oxides of nitrogen, and gaseous fluorides emitted during conventional processing or vitrification of fuel Group B. The concentrations would range from less than 1 percent to about 55 percent of the offsite standard (for nitrogen oxides). If a new facility or a major modification to an existing facility were being considered, new permitting actions would be required as part of the Clean Air Act Title V permit compliance requirements. Under the current Title V permit, SRS would have to conduct a Prevention of Significant Deterioration review, since the nitrogen oxide levels exceed the 25 μm per cubic meter per year threshold of NO_2 for a Class II area. In addition, there would be a requirement for ambient monitoring to verify emission levels once the process began.

Offsite Concentrations

This analysis presents projected maximum offsite nonradiological incremental air concentrations in much the same way it presents the onsite concentrations. The estimated maximum incremental concentrations listed in Tables F-6 through F-10 in Appendix F would occur at the SRS boundary for emissions associated with SNF. The air quality regulatory standards listed in the tables are applicable to the Site boundary concentrations from all SRS emissions. All the

incremental concentrations are at least three orders of magnitude less than any of the corresponding standards except those for oxides of nitrogen and gaseous fluorides emitted during conventional processing or vitrification. The concentrations ranged from less than 1 percent to about 2 percent of the offsite standard.

4.1.1.2.2 Radiological Emissions

DOE estimated airborne radionuclide emission rates for each technology option (Bickford et al. 1997), and used them with the meteorology data from Section 3.3.1 as inputs to the SRS computer models MAXIGASP and POPGASP (Hamby 1994) to determine doses to onsite (noninvolved worker) and offsite (hypothetical maximally exposed individual) recipients and the surrounding population (620,000 persons) within a 50-mile (80-kilometer) radius of the center of the Site (Simpkins 1996). The analysis uses the meteorological data to determine annual average concentrations in air. The values presented in Tables 4.1-3, 4.1-4, and 4.1-5 represent current reactor-area emissions (including two SNF wet basins).

Onsite Doses

Atmospheric doses to the noninvolved worker represent the radiological exposures of a hypothetical worker who is nearby but not involved in SNF operations. Table 4.1-3 lists the estimated maximum incremental annual doses to noninvolved workers from atmospheric emissions of radionuclides for each viable technology option for each fuel group. The EPA limit of 10 millirem per year (40 CFR Part 61, Subpart H) is a point of comparison for these doses. (In fact, this limit is applicable to offsite individuals from sitewide airborne releases; see Chapter 5). The highest incremental dose to the noninvolved worker would be 0.27 millirem (from Melt and Dilute, Vitrification, or Electrometallurgical Treatment of Materials Test Reactor-like Fuels). Incremental doses to the noninvolved worker from all viable options would be 3 percent or less of the national emission standards for hazardous air pollutants (NESHAP) limit.

There would be no pathways for exposure of personnel inside SNF management facilities from atmospheric releases of radioactivity. Section 4.1.1.3 discusses radiation doses to SNF management workers, including from in-facility airborne releases of radioactivity.

Offsite Doses

Atmospheric doses to the hypothetical maximally exposed offsite individual assume a person who resides at the SRS boundary at the point of maximum exposure. Every member of the public would have a dose less than that received by this individual. Table 4.1-4 lists the estimated maximum incremental annual dose to this individual from atmospheric emissions of radionuclides for each technology option for each fuel group. As with the doses to noninvolved workers, the NESHAP limit of 10 millirem per year (40 CFR Part 61, Subpart H) is a point of comparison. The maximum incremental annual dose from any technology option for a given fuel group would be 0.033 millirem per year (from Melt and Dilute, Vitrification, or Electrometallurgical Treatment of Materials Test Reactor-like Fuels), a factor of 300 less than the EPA limit.

Table 4.1-5 lists the estimated maximum incremental annual population dose (the collective dose to the entire population around SRS) for each viable option. The maximum incremental annual population dose from any option would be 1.2 person-rem per year (from Melt and Dilute, Vitrification, or Electrometallurgical Treatment of Materials Test Reactor-like Fuels).

4.1.1.3 Worker and Public Health

This section discusses potential radiological and nonradiological health effects to SRS workers and the surrounding public from the technology options for the management of SNF; it does not include impacts of potential accidents, which are discussed in Section 4.2. DOE based its calculations of health effects from the air- and waterborne radiological releases on (1) the dose to the hypothetical maximally exposed individual

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Table 4.1-3. Estimated maximum incremental annual dose (millirem) to noninvolved worker from airborne releases.

Fuel group	Technologies							
	1	2	3	4	5	6	7	8
	Prepare for Repackage direct and prepare co-disposal to ship							
A. Uranium and Thorium Metal Fuels	0 ^b	NA	5.3×10 ⁻⁴	NA	5.3×10 ⁻⁴	5.3×10 ⁻⁴	3.2×10 ⁻⁴	1.8×10 ^{-3c}
B. Materials Test Reactor-Like Fuels	0 ^b	NA	0.27	0.013	0.27	0.27	0.09	0.083 ^c
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	0 ^b	NA	0.085	0.0043	0.085	0.085	0.029	0.02 ^c
D. Loose Uranium Oxide in Cans	NA	NA	5.0×10 ⁻³	NA	5.0×10 ⁻³	5.0×10 ⁻³	5.7×10 ⁻³	4.7×10 ^{-3c}
E. Higher Actinide Targets	NA	0 ^b	NA	NA	NA	NA	NA	6.7×10 ^{-4c}
F. Non-Aluminum-Clad Fuels	NA	0 ^b	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.

HEU = Highly Enriched Uranium.

LEU = Low Enriched Uranium.

- a. Annual impacts from Conventional Processing are lower because the amount of material processed annually by this technology is less than for other technologies. The annual impacts for Conventional Processing are based on operating one dissolver in a canyon. Impacts would double if the canyon was operated at full capacity (i.e., two dissolvers). Fuel processing of the entire SNF inventory would take over 20 dissolver-years using one dissolver and about 11 dissolver-years using two dissolvers. Processing all the fuel at full capacity in a new treatment facility would take about 7 years. Appendix E provides more information related to processing durations.
- b. No incremental increase expected above SRS baseline radioactive emissions values reported in Chapter 3 because these options would not change the integrity of the fuel.
- c. Reflects current reactor-area emissions (including two SNF wet basins).

Table 4.1-4. Estimated maximum incremental annual dose (millirem) to hypothetical maximally exposed offsite individual from airborne releases.

Fuel group	Technologies							
	1	2	3	4	5	6	7	8
	Prepare for Repackage							
	direct co-disposal	and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing ^a	Continued wet storage
A. Uranium and Thorium Metal Fuels	0 ^b	NA	6.5×10 ⁻⁵	NA	6.5×10 ⁻⁵	6.5×10 ⁻⁵	3.9×10 ⁻⁵	2.6×10 ^{-4c}
B. Materials Test Reactor-Like Fuels	0 ^b	NA	0.033	0.0016	0.033	0.033	0.011	0.012 ^c
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	0 ^b	NA	0.010	5.2×10 ⁻⁴	0.010	0.010	3.5×10 ⁻³	3.3×10 ^{-3c}
D. Loose Uranium Oxide in Cans	NA	NA	6.1×10 ⁻⁴	NA	6.1×10 ⁻⁴	6.1×10 ⁻⁴	7.0×10 ⁻⁴	6.9×10 ^{-4c}
E. Higher Actinide Targets	NA	0 ^b	NA	NA	NA	NA	NA	9.9×10 ^{-5c}
F. Non-Aluminum-Clad Fuels	NA	0 ^b	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.

HEU = Highly Enriched Uranium.

LEU = Low Enriched Uranium.

- a. Annual impacts from Conventional Processing are lower because the amount of material processed annually by this technology is less than for other technologies. The annual impacts for Conventional Processing are based on operating one dissolver in a canyon. Impacts would double if the canyon was operated at full capacity (i.e., two dissolvers). Fuel processing of the entire SNF inventory would take over 20 dissolver-years using one dissolver and about 11 dissolver-years using two dissolvers. Processing all the fuel at full capacity in a new treatment facility would take about 7 years. Appendix E provides more information related to processing durations.
- b. No incremental increase expected above SRS baseline radioactive emissions values reported in Chapter 3 because these options would not change the integrity of the fuel.
- c. Reflects current reactor-area emissions (including two SNF wet basins).

Table 4.1-5. Estimated maximum incremental annual dose (person-rem) to the 620,100 person population surrounding SRS from airborne releases.

	Technologies								
	1	2	3	4	5	6	7	8	
Fuel group	Prepare for Repackage direct and prepare co-disposal		to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing ^a	Continued wet storage
A. Uranium and Thorium Metal Fuels	0 ^b	NA	2.4×10 ⁻³	NA	2.4×10 ⁻³	2.4×10 ⁻³	1.4×10 ⁻³	9.5×10 ^{-3c}	
B. Materials Test Reactor-Like Fuels	0 ^b	NA	1.2	0.060	1.2	1.2	0.41	0.44 ^c	
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	0 ^b	NA	0.38	0.019	0.38	0.38	0.13	0.12 ^c	
D. Loose Uranium Oxide in Cans	NA	NA	0.022	NA	0.022	0.022	0.026	0.025 ^c	
E. Higher Actinide Targets	NA	0 ^b	NA	NA	NA	NA	NA	3.57×10 ^{-3c}	
F. Non-Aluminum-Clad Fuels	NA	0 ^b	NA	NA	NA	NA	NA	NA	NA

in the public; (2) the collective dose to the population within a 50-mile (80-kilometer) radius around the SRS (approximately 620,000 people); (3) the collective dose to workers involved in implementing a given alternative (i.e., the workers involved in SNF management activities); and (4) the dose to the maximally exposed noninvolved worker (i.e., SRS employees who may work in the vicinity of the SNF management facilities but are not directly involved in SNF work). All radiation doses mentioned in this EIS are effective dose equivalents; internal exposures are committed effective dose equivalents. This section presents total impacts for the entire length of time necessary to implement each technology, using the durations listed in Appendix E. The annual impacts attributable to each phase were multiplied by the duration of that phase. The impacts from all phases were summed to calculate the total impact for the technology. This discussion characterizes health effects as additional lifetime latent cancer fatalities likely to occur in the general population around SRS and in the population of workers who would be associated with the options.

4.1.1.3.1 Radiological Health Effects

Radiation can cause a variety of health effects in people. The major effects that environmental and occupational radiation exposures could cause are delayed cancer fatalities, which are called latent cancer fatalities because the cancer can take many years to develop and cause death.

To relate a dose to its effect, DOE has adopted a dose-to-risk conversion factor of 0.0004 latent cancer fatality per person-rem for workers and 0.0005 latent cancer fatality per person-rem for the general population (NCRP 1993). The factor for the population is slightly higher due to the presence of infants and children who might be more sensitive to radiation than workers, who are, generally speaking, healthy adults.

DOE uses these conversion factors to estimate the effects of exposing a population to radiation. For example, in a population of 100,000 people exposed only to background radiation (0.3 rem per year), DOE would calculate 15 latent cancer

fatalities per year caused by radiation ($100,000 \text{ persons} \times 0.3 \text{ rem per year} \times 0.0005 \text{ latent cancer fatality per person-rem}$).

Calculations of the number of latent cancer fatalities associated with radiation exposure might not yield whole numbers and, especially in environmental applications, might yield values less than 1. For example, if a population of 100,000 were exposed only to a dose of 0.001 rem to each person, the collective dose would be 100 person-rem, and the corresponding number of latent cancer fatalities would be 0.05 ($100,000 \text{ persons} \times 0.001 \text{ rem} \times 0.0005 \text{ latent cancer fatality per person-rem}$).

DOE also has employed these concepts in estimating the effects of radiation exposure to a single individual. For example, consider the effects of exposure to background radiation over a lifetime. The number of latent cancer fatalities corresponding to an individual's exposure over a (presumed) 72-year lifetime at 0.3 rem per year would be 0.011 latent cancer fatality ($1 \text{ person} \times 0.3 \text{ rem per year} \times 72 \text{ years} \times 0.0005 \text{ latent cancer fatality per person-rem}$).

This number should be interpreted in a statistical sense; that is, the estimated effect of background radiation exposure to the exposed individual is a 1.1-percent lifetime chance that the individual might incur a latent fatal cancer. Vital statistics on mortality rates for 1994 (CDC 1996) indicate that the overall lifetime fatality rate in the United States from all forms of cancer is about 23.4 percent (23,400 fatal cancers per 100,000 deaths).

These factors, which DOE uses in this EIS to relate radiation exposure to latent cancer fatalities, are based on the *Recommendations of the International Commission on Radiation Protection* (ICRP 1991). They are consistent with the factors used by the U.S. Nuclear Regulatory Commission in its rulemaking *Standards for Protection Against Radiation* (10 CFR Part 20). The factors apply if the dose to an individual is less than 20 rem and the dose rate is less than 10 rem per hour. At doses greater than 20 rem, the factors used to relate radiation doses to latent

cancer fatalities are doubled. At much higher dose rates, prompt effects, rather than latent cancer fatalities, would be the primary concern.

In addition to latent cancer fatalities, other health effects could result from environmental and occupational exposures to radiation; these include nonfatal cancers among the exposed population and genetic effects in subsequent generations. Previous studies have concluded that these effects are less probable than fatal cancers as consequences of radiation exposure (ICRP 1991). Dose-to-risk conversion factors for nonfatal cancers and hereditary genetic effects (0.0001 per person-rem and 0.00013 per person-rem, respectively) are substantially lower than those for fatal cancers. This EIS presents estimated effects of radiation only in terms of latent cancer fatalities because that is the major potential health effect from exposure to radiation. Estimates of nonfatal cancers and hereditary genetic effects can be estimated by multiplying the radiation doses by the effects dose-to-risk conversion factors.

DOE expects minimal worker and public health impacts from the radiological consequences of managing SNF under any of the technology options, as well as Continued Wet Storage. However, some options would result in increased radiological releases. Public radiation doses include doses from airborne releases (Section 4.1.1.2) and liquid releases (Section 4.1.1.1). Table 4.1-6 lists incremental radiation doses estimated for the public (maximally exposed individual and collective population dose) and corresponding incremental latent cancer fatalities, for each fuel group and technology option.

The values in Tables 4.1-6 and 4.1-8 for the No-Action Alternative represent current reactor-area emissions (including two SNF wet basins) for the entire period of analysis. The values for the other alternatives would be incremental above these baseline values. Summing these baseline and incremental values would be conservative, however, because there would not be two SNF wet basins operating over the entire 38-year period of analysis.

DOE based estimated worker doses on past operating experience and the projected durations for implementation of the alternative actions (Bickford et al. 1997). For the maximally exposed worker, DOE assumed that no worker would receive an annual dose greater than 500 millirem from any option because SRS uses the 500-millirem value as an administrative limit for normal operations; that is, an employee who receives an annual dose approaching the administrative limit normally is reassigned to duties in a nonradiation area. (Note: If DOE privatized the Transfer and Storage Facility or treatment operations, the licensee would adopt NRC worker dose limits, and administrative limits could be subject to adjustment.) Tables 4.1-7 and 4.1-8 estimate radiation doses for the collective population of workers who would be directly involved in implementing the options and for the noninvolved worker (a worker not directly involved with implementing the option but located 2,100 feet [640 meters] from the SNF facility) for each fuel group and technology option. These tables also list the latent cancer fatalities likely attributable to the doses.

Of the fuels considered for treatment (all except higher actinide targets and non-aluminum clad fuel), the highest expected radiological health effects to the public generally would occur under conventional processing. The single exception would be fewer latent cancer fatalities predicted for the population from the conventional processing of uranium and thorium metal fuels (Table 4.1-6). For the noninvolved workers, the conventional processing of Groups C and D fuels would result in the greatest radiological health effects. No measurable incremental increases would be likely for the higher actinide targets or the non-aluminum-clad fuels for any option because the only options applied to those groups are repackaging and continued wet storage. The estimated collective dose for workers who would be directly involved in managing SNF (Table 4.1-7) depends largely on the difference in the number of workers involved in each option and not on the difference in the amount of radioactivity.

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Table 4.1-6. Radiation doses to the public and associated latent cancer fatalities for the entire period of analysis (1998-2035).^a

Technologies										
		1	2	3	4	5	6	7	8	
		Prepare for direct co-disposal		Repackage and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing	Continued wet storage
Fuel Group	Parameter									
A. Uranium and Thorium Metal Fuels	MEI ^b dose (millirem)	0 ^c	NA	6.5×10 ⁻⁵	NA	NA	6.5×10 ⁻⁵	6.5×10 ⁻⁵	7.3×10 ⁻⁵	0.01 ^g
	MEI LCF ^{d,e}	0 ^c	NA	3.2×10 ⁻¹¹	NA	NA	3.2×10 ⁻¹¹	3.2×10 ⁻¹¹	3.6×10 ⁻¹¹	5.0×10 ^{-6g}
	Collective population dose (person-rem)	0 ^c	NA	2.4×10 ⁻³	NA	NA	2.4×10 ⁻³	2.4×10 ⁻³	1.6×10 ⁻³	0.36 ^g
	Collective population LCF ^f	0 ^c	NA	1.2×10 ⁻⁶	NA	NA	1.2×10 ⁻⁶	1.2×10 ⁻⁶	8.1×10 ⁻⁷	1.8×10 ^{-4g}
B. Materials Test Reactor-Like Fuels	MEI ^b dose (millirem)	0 ^c	NA	0.17	9.0×10 ⁻⁵	0.17	0.17	0.17	0.54	0.46 ^g
	MEI LCF ^{d,e}	0 ^c	NA	8.5×10 ⁻⁸	4.5×10 ⁻⁹	8.5×10 ⁻⁸	8.5×10 ⁻⁸	8.5×10 ⁻⁸	2.7×10 ⁻⁷	2.3×10 ^{-4g}
	Collective population dose (person-rem)	0 ^c	NA	6.3	0.3	6.3	6.3	6.3	7.3	16.7 ^g
	Collective population LCF ^f	0 ^c	NA	3.1×10 ⁻³	1.7×10 ⁻⁴	3.1×10 ⁻³	3.1×10 ⁻³	3.1×10 ⁻³	3.7×10 ⁻³	8.3×10 ^{-3g}
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	MEI ^b dose (millirem)	0 ^c	NA	0.015	7.8×10 ⁻⁴	0.015	0.015	0.015	0.12	0.12 ^g
	MEI LCF ^{d,e}	0 ^c	NA	7.3×10 ⁻⁹	3.9×10 ⁻¹⁰	7.3×10 ⁻⁹	7.3×10 ⁻⁹	7.3×10 ⁻⁹	6.2×10 ⁻⁸	6.2×10 ^{-5g}
	Collective population dose (person-rem)	0 ^c	NA	0.54	0.029	0.54	0.54	0.54	1.3	4.5 ^g
	Collective population LCF ^f	0 ^c	NA	2.7×10 ⁻⁴	1.4×10 ⁻⁵	2.7×10 ⁻⁴	2.7×10 ⁻⁴	2.7×10 ⁻⁴	6.5×10 ⁻⁴	2.2×10 ^{-3g}
D. Loose Uranium Oxide in Cans	MEI ^b dose (millirem)	NA	NA	6.1×10 ⁻⁴	NA	NA	6.1×10 ⁻⁴	6.1×10 ⁻⁴	7.1×10 ⁻³	0.026 ^g
	MEI LCF ^{d,e}	NA	NA	3.0×10 ⁻¹⁰	NA	NA	3.0×10 ⁻¹⁰	3.0×10 ⁻¹⁰	3.6×10 ⁻⁹	1.3×10 ^{-5g}
	Collective population dose (person-rem)	NA	NA	0.022	NA	NA	0.022	0.022	0.075	0.95 ^g
	Collective population LCF ^f	NA	NA	1.1×10 ⁻⁵	NA	NA	1.1×10 ⁻⁵	1.1×10 ⁻⁵	3.8×10 ⁻⁵	4.7×10 ^{-4g}
E. Higher Actinide Targets	MEI ^b dose (millirem)	NA	0 ^c	NA	NA	NA	NA	NA	NA	3.7×10 ^{-3g}
	MEI LCF ^{d,e}	NA	0 ^c	NA	NA	NA	NA	NA	NA	1.9×10 ^{-6g}
	Collective population dose (person-rem)	NA	0 ^c	NA	NA	NA	NA	NA	NA	0.14 ^g
	Collective population LCF ^f	NA	0 ^c	NA	NA	NA	NA	NA	NA	6.8×10 ^{-5g}
F. Non-Aluminum-Clad Fuels	MEI ^b dose (millirem)	NA	0 ^c	NA	NA	NA	NA	NA	NA	NA
	MEI LCF ^{d,e}	NA	0 ^c	NA	NA	NA	NA	NA	NA	NA
	Collective population dose (person-rem)	NA	0 ^c	NA	NA	NA	NA	NA	NA	NA
	Collective population LCF ^f	NA	0 ^c	NA	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.

HEU = Highly Enriched Uranium.

LEU = Low Enriched Uranium.

- a. Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.
- b. MEI = Maximally Exposed Individual; i.e., a hypothetical member of the public whose location and habits result in exposure to the maximum dose from all pathways.
- c. No incremental increase expected above SRS baseline radioactive emissions values presented in Chapter 3 because these options would not affect the integrity of the fuel.
- d. LCF = latent cancer fatalities.
- e. For an individual, the LCF value should be interpreted statistically; e.g., 1×10^{-9} = 1 chance in 1 billion to develop a fatal cancer.
- f. For collective population, the LCF value should be interpreted as the number of cancers that could be expected in the population.
- g. Reflects current reactor-area emissions (including two SNF wet basins) for the entire period of analysis.

Table 4.1-7. Number of radiation workers and collective worker radiation dose (person-rem) and associated latent cancer fatalities for the entire period of analysis (1998-2035).^a

Fuel Group	Parameter	Technologies							
		1	2	3	4	5	6	7	8
	Number of radiation workers ^b	75	38	100	88	159	119	150	40
	Prepare for direct co-disposal								
	Repackage and prepare to ship								
	Melt and dilute								
	Mechanical dilution								
	Virification technologies								
	Electrometallurgical treatment								
	Conventional processing								
	Continued wet storage								
A. Uranium and Thorium Metal Fuels	Collective worker dose (person-rem)	11	NA	12	NA	15	13	18	12
	LCF ^c	4.2×10 ⁻³	NA	4.8×10 ⁻³	NA	6.1×10 ⁻³	5.2×10 ⁻³	7.2×10 ⁻³	4.9×10 ⁻³
B. Materials Test Reactor-Like Fuels	Collective worker dose (person-rem)	480	NA	530	520	680	580	1,300	560
	LCF	0.19	NA	0.21	0.21	0.27	0.23	0.50	0.22
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	Collective worker dose (person-rem)	140	NA	150	150	190	160	600	150
	LCF	0.054	NA	0.059	0.059	0.075	0.064	0.24	0.060
D. Loose Uranium Oxide in Cans	Collective worker dose (person-rem)	NA	NA	31	NA	40	34	170	32
	LCF	NA	NA	0.012	NA	0.016	0.014	0.069	0.013
E. Higher Actinide Targets	Collective worker dose (person-rem)	NA	3	NA	NA	NA	NA	NA	5
	LCF	NA	1.3×10 ⁻³	NA	NA	NA	NA	NA	1.8×10 ⁻³
F. Non-Aluminum Clad Fuels	Collective worker dose (person-rem)	NA	26	NA	NA	NA	NA	NA	NA
	LCF	NA	0.011	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.

HEU = Highly Enriched Uranium.

LEU = Low Enriched Uranium.

a. Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.

b. Estimates of the number of radiation workers are based on past operating experience (Bickford et al. 1997).

c. LCF = latent cancer fatalities.

Table 4.1-8. Radiation doses to the maximally exposed noninvolved worker (at 640 meters) and associated latent cancer fatalities for the entire period of analysis (1998-2035).^a

Fuel Group	Parameter	Technologies							
		1	2	3	4	5	6	7	8
		Prepare for direct co-disposal	Repackage and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing	Continued wet storage
A. Uranium and Thorium Metal Fuels	Noninvolved worker dose (millirem)	0 ^c	NA	5.3×10 ⁻⁴	NA	5.3×10 ⁻⁴	5.3×10 ⁻⁴	3.2×10 ⁻⁴	0.068 ^d
	Noninvolved worker LCF ^b	0 ^c	NA	2.1×10 ⁻¹⁰	NA	2.1×10 ⁻¹⁰	2.1×10 ⁻¹⁰	1.3×10 ⁻¹⁰	2.7×10 ^{-5d}
B. Materials Test Reactor-Like Fuels	Noninvolved worker dose (millirem)	0 ^c	NA	1.4	0.074	1.4	1.4	1.3	3.1 ^d
	Noninvolved worker LCF ^b	0 ^c	NA	5.6×10 ⁻⁷	2.9×10 ⁻⁸	5.6×10 ⁻⁷	5.6×10 ⁻⁷	5.4×10 ⁻⁷	1.3×10 ^{-3d}
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	Noninvolved worker dose (millirem)	0 ^c	NA	0.12	6.3×10 ⁻³	0.12	0.12	0.22	0.84 ^d
	Noninvolved worker LCF ^b	0 ^c	NA	4.8×10 ⁻⁸	2.5×10 ⁻⁹	4.8×10 ⁻⁸	4.8×10 ⁻⁸	8.6×10 ⁻⁸	3.4×10 ^{-4d}
D. Loose Uranium Oxide in Cans	Noninvolved worker dose (millirem)	NA	NA	5.0×10 ⁻³	NA	5.0×10 ⁻³	5.0×10 ⁻³	0.013	0.18 ^d
	Noninvolved worker LCF ^b	NA	NA	2.0×10 ⁻⁹	NA	2.0×10 ⁻⁹	2.0×10 ⁻⁹	5.0×10 ⁻⁹	7.1×10 ^{-5d}
E. Higher Actinide Targets	Noninvolved worker dose (millirem)	NA	0 ^c	NA	NA	NA	NA	NA	0.025 ^d
	Noninvolved worker LCF ^b	NA	0 ^c	NA	NA	NA	NA	NA	1.0×10 ^{-5d}
F. Non-Aluminum-Clad Fuels	Noninvolved worker dose (millirem)	NA	0 ^c	NA	NA	NA	NA	NA	NA
	Noninvolved worker LCF ^b	NA	0 ^c	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.

HEU = Highly Enriched Uranium.

LEU = Low Enriched Uranium.

- Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.
- LCF = latent cancer fatalities; this number should be interpreted statistically.
- No incremental increase expected above SRS baseline radioactive emissions values presented in Chapter 3, because these options would not affect the integrity of the fuel.
- Reflects current reactor-area emissions (including two SNF wet basins) for the entire period of analysis.

The estimated number of latent cancer fatalities in the public listed in Table 4.1-6 can be compared to the projected number of fatal cancers (145,100) in the public around the SRS from all causes (as discussed in Section 3.7.1). Similarly, the estimated number of latent cancer fatalities in the worker population can be compared to the number in the worker population from all causes (approximately 23.4 percent; see Section 3.7.1). In all cases, the incremental impacts from the options would be negligible.

4.1.1.3.2 Nonradiological Health Effects

DOE evaluated the range of chemicals to which the public and workers would be exposed due to SNF management activities and expects minimal health impacts from nonradiological exposures. Section 4.1.1.1.1 discusses offsite chemical concentrations from air emissions. DOE estimated worker impacts and compared them to Occupational Safety and Health Administration (OSHA) permissible exposure limits (PELs) or ceiling limits for protecting worker health, and concluded that all impacts would be well below the limits.

OSHA limits (29 CFR Part 1910.1000) are time-weighted average concentrations that a facility cannot exceed during a prescribed duration of a 40-hour week. The facility cannot exceed OSHA ceiling concentrations during any part of the workday. These exposure limits refer to airborne concentrations of substances and repre-

sent conditions under which nearly all workers could be exposed day after day without adverse health effects. However, because of the wide variation in individual susceptibility, a small percentage of workers could experience discomfort from some substances at concentrations at or below the permissible limit. Table 4.1-9 summarizes the values of Permissible Exposure Limits that DOE compared to the data in Tables F-1 through F-5 in Appendix F.

4.1.1.4 Waste Generation

This section presents waste generation estimates for each technology option and fuel group that DOE considers in this EIS. Tables 4.1-10 through 4.1-13 list these estimates. For each technology option, this analysis considered three handling phases as potential sources of waste: wet storage (pretreatment storage), treatment or conditioning, and dry storage (post-treatment storage pending final disposition). The period and waste generation rate associated with each phase varied depending on the fuel group and the technology. As discussed above, DOE summed waste volumes from each phase; the values listed in the tables represent the total projected waste volumes for each technology option in a given fuel group.

DOE used the annual waste generation rates to calculate the estimates in the tables (Bickford et al. 1997); the rates are based on applicable current and past SRS operations or on process

Table 4.1-9. Permissible Exposure Limits (milligrams per cubic meter) of nonradiological air pollutants regulated by the Occupational Safety and Health Administration.^a

Pollutant	Averaging time	OSHA PEL ^b
Carbon monoxide	8 hours	55
Nitrogen oxides	1 hour	9 ^c
Sulfur dioxide	8 hours	13
Carbon dioxide	8 hours	9,000
Nitric acid	8 hours	5

a. Source: 29 CFR Part 1910.1000.

b. Occupational Safety and Health Administration (OSHA) permissible exposure limit (PEL).

c. OSHA ceiling limit not to be exceeded at any time during the workday.

Table 4.1-10. High-level waste generation for the entire period of analysis (1998-2035) (cubic meters).^{a,b,c}

Fuel group	Parameter	Technologies							
		1	2	3	4	5	6	7	8
		Prepare for direct co-disposal	Repackage and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing	Continued Wet storage
A. Uranium and Thorium Metal Fuels	Liquid high-level waste Equivalent DWPF canisters	10	NA	10	NA	10	10	170	36
	Saltstone	<1	NA	<1	NA	<1	<1	3	<1
		26	NA	26	NA	26	26	430	97
B. Materials Test Reactor-Like Fuels	Liquid high-level waste Equivalent DWPF canisters	470	NA	450	470	450	450	7,700	1,700
	Saltstone	8	NA	7	8	7	7	120	28
		1,250	NA	1,200	1,300	1,200	1,200	20,000	4,500
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	Liquid high-level waste Equivalent DWPF canisters	125	NA	120	130	120	120	2,100	450
	Saltstone	2	NA	2	2	2	2	32	8
		330	NA	320	340	320	320	5,400	1,200
D. Loose Uranium Oxide in Cans	Liquid high-level waste Equivalent DWPF canisters	NA	NA	25	NA	25	25	450	96
	Saltstone	NA	NA	<1	NA	<1	<1	7	2
		NA	NA	67	NA	67	67	1,100	260
E. Higher Actinide Targets	Liquid high-level waste Equivalent DWPF canisters	NA	4	NA	NA	NA	NA	NA	14
	Saltstone	NA	<1	NA	NA	NA	NA	NA	<1
		NA	10	NA	NA	NA	NA	NA	36
F. Non-Aluminum-Clad Fuels	Liquid high-level waste Equivalent DWPF canisters	NA	30	NA	NA	NA	NA	NA	NA
	Saltstone	NA	<1	NA	NA	NA	NA	NA	NA
		NA	80	NA	NA	NA	NA	NA	NA

DWPF = Defense Waste Processing Facility.

NA = Technology is not applicable to this fuel type.

HEU = Highly Enriched Uranium.

LEU = Low Enriched Uranium.

a. Except DWPF canisters.

b. To convert cubic meters to cubic yards, multiply by 1.308.

c. Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.

Table 4.1-11. Transuranic waste generation for the entire period of analysis (1998-2035) (cubic meters).^a

Fuel group	Technologies									
	1	2	3	4	5	6	7	8		
	Prepare for direct co-disposal		Repackage and prepare to ship		Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing	Continued wet storage
A. Uranium and Thorium Metal Fuels	0	NA	5.5	NA	11	11	30	0		
B. Materials Test Reactor-Like Fuels	0	NA	260	415	520	520	2,200	0		
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	0	NA	70	110	140	140	1,100	0		
D. Loose Uranium Oxide in Cans	NA	NA	15	NA	29	29	330	0		
E. Higher Actinide Targets	NA	0	NA	NA	NA	NA	NA	0		
F. Non-Aluminum-Clad Fuels	NA	0	NA	NA	NA	NA	NA	NA		NA

Table 4.1-13. Low-level waste generation for the entire period of analysis (1998-2035) (cubic meters).^a

Fuel group	Technologies							
	1	2	3	4	5	6	7	8
	Prepare for direct co-disposal	Repackage and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing	Continued wet storage
A. Uranium and Thorium Metal Fuels	290	NA	440	NA	660	660	1,200	910
B. Materials Test Reactor-Like Fuels	14,000	NA	20,000	21,000	31,000	31,000	85,000	42,000
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	3,700	NA	5,500	5,800	8,200	8,200	40,000	11,000
D. Loose Uranium Oxide in Cans	NA	NA	1,150	NA	1,700	1,700	12,000	2,400
E. Higher Actinide Targets	NA	110	NA	NA	NA	NA	NA	340
F. Non-Aluminum-Clad Fuels	NA	900	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.

HEU = Highly Enriched Uranium.

LEU = Low Enriched Uranium.

a. Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.

knowledge for new treatment technologies. The operating history that was the basis for these estimates would maximize projected waste generation rates. As described in Section 3.8, the Site generates several types of waste (high-level, transuranic, mixed, hazardous, low-level, and sanitary). Wastes generated by SNF management activities would be comparable to wastes the SRS currently handles and would, therefore, not require unique treatment, storage, or disposal actions. This section does not consider sanitary waste, the production of which would be in direct proportion to the number of employees, because none of the technologies would increase the number of permanent Site employees.

DOE has implemented an aggressive waste minimization and pollution prevention program at SRS at the sitewide level and for individual organizations and projects. As a result, significant reductions have been achieved in the amounts of wastes discharged into the environment and sent to landfills, resulting in significant cost savings.

To implement a waste minimization and pollution prevention program at the SNF management facilities, DOE would characterize waste streams and identify opportunities for reducing or eliminating them. Emphasis would be placed on minimizing the largest waste stream, low-level waste, through source reduction and recycling. Selected waste minimization practices could include: (1) process design changes to reduce the potential for spills and to minimize contamination areas, (2) decontamination of equipment to facilitate reuse, (3) recycling metals and other usable materials, especially during the construction phase of the project, (4) preventive maintenance to extend process equipment life, (5) modular equipment designs to isolate potential failure elements to avoid changing out entire units, and (6) use of non-toxic or less toxic materials to prevent pollution and minimize hazardous and mixed waste streams.

The following sections describe the differences in waste generation by waste type among the

SNF management technologies considered in this EIS.

4.1.1.4.1 High-Level Waste

SRS reports high-level waste as liquid high-level waste, and in the related quantities of equivalent Defense Waste Processing Facility (DWPF) canisters and saltstone. The volume estimates for liquid high-level waste reported in Table 4.1-10 are for volumes as they leave the process and enter the high-level waste tanks. While it is necessary to consider this volume when evaluating the interim storage of high-level waste in the tank farms, the volume of liquid high-level waste is not meaningful when considering the storage and disposition of final waste forms. The liquid waste is evaporated and concentrated in the high-level waste tanks. The generation of secondary waste in the high-level waste tanks and DWPF, including waste generated as a result of activities described in this SNF EIS, is evaluated in the DWPF Supplemental EIS (DOE 1994). Therefore, capacity for management of SNF secondary waste in the tank farms and DWPF is provided within the scope of DWPF operations. DWPF canisters and saltstone are the product of liquid high-level waste treatment and evaporation and would be the basis for final storage and disposition considerations. Because the production of saltstone and DWPF canisters from a given liquid waste volume are generally proportional, this discussion applies equally to DWPF canisters and saltstone. For Conventional Processing, DWPF canisters would be the only product to be disposed in a geologic repository.

Conventional Processing is the only option that would generate significant quantities of high-level waste during the treatment phase. Each option would produce high-level waste during the wet storage phase and technologies such as melt and dilute, that require off-gas collection systems, would also produce high-level waste, but the quantity produced generally would be much lower than that associated with Conventional Processing. The waste generated during wet storage and new technology processing operations would not meet the formal definition of

high-level waste (waste resulting from the processing of SNF), but would consist of such items as deionizer backwash and off-gas collection products, which the SRS typically manages (or would manage) as high-level waste. The lengthy period associated with continued wet storage generally would make it the second largest producer of high-level waste. For the higher actinide targets, Conventional Processing was not considered, making Continued Wet Storage the greatest potential for high-level waste production. The volumes of high-level waste generated by the other options would vary depending on the duration of storage and the amount of fissile material in the fuel, but would be fairly comparable within a given fuel type and substantially less than the volumes associated with conventional processing. In addition, the condition of the fuel would influence the high-level waste generation rate (i.e., fuel in poor condition would result in higher generation of deionizer backwash).

Based on the capacities of the high-level waste tank farms and the current volume of high-level waste in storage (see Table 3.8-2), these projected high-level waste volumes probably would not require additional treatment and storage facilities beyond those currently available at SRS. DOE bases this conclusion on continued removal and treatment of the existing tank farm inventory. DWPF would be available to treat these projected high-level waste volumes.

4.1.1.4.2 Transuranic Waste

For all applicable fuel types, conventional processing would produce the largest volume of transuranic waste due to a higher generation rate and a longer processing time. Conventional processing of all applicable fuel groups would generate 3660 cubic meters of transuranic waste which is 29 percent of the total SRS transuranic waste generation forecast (Table 3.8-1). The next largest quantity that could be generated would be from the Vitrification and Electrometallurgical Treatments of all applicable fuel groups. Those technologies would generate 700 cubic meters of transuranic waste over the life of the project, which is less than 6 percent of the

total SRS transuranic waste generation forecast. These two technologies would produce 9 to 37 percent of that produced by conventional processing, depending on the fuel group.

None of the treatment options associated with the higher actinide targets or non-aluminum-clad fuels would produce transuranic waste.

4.1.1.4.3 Hazardous/Low-Level Mixed Waste

For this EIS analysis, DOE grouped hazardous and low-level mixed wastes together because none of the options is likely to produce significant quantities of either.

The highest hazardous/low-level mixed waste generation rates would be associated with Vitrification and Electrometallurgical Treatments, followed by Mechanical Dilution. However, due to the longer time required to process the loose uranium oxide in cans, the Materials Test Reactor-like fuels, and the highly enriched uranium/low enriched uranium (HEU/LEU) oxides and silicides requiring resizing or special packaging, conventional processing would produce the largest volume of hazardous or mixed waste for those fuel groups. Vitrification and Electrometallurgical Treatments generally would produce the next largest quantities (35 to 88 percent of that produced by conventional processing, depending on the fuel group). For the uranium and thorium metal fuels, Vitrification and Electrometallurgical Treatments produce the largest quantities of hazardous/low-level mixed waste, followed by conventional processing. For applicable fuel groups, the Direct Disposal/Direct Co-Disposal technology would consistently produce the smallest quantities of hazardous or mixed waste. The waste volumes that continued wet storage or the Melt and Dilute technology would produce would be roughly comparable and generally intermediate among the technologies. For the higher actinide targets, the two technologies being considered (Repackage and Prepare to Ship and Continued Wet Storage) would produce small, comparable quantities of hazardous or mixed waste.

When all applicable technologies are considered, conventional processing would generate the largest volume (264 cubic meters) of hazardous and low-level mixed waste, which is less than 1 percent of the 30-year forecast.

4.1.1.4.4 Low-Level Waste

The Direct Disposal/Direct Co-Disposal and Repackage and Prepare to Ship technology options would produce the least low-level waste. The Mechanical Dilution and Melt and Dilute options would produce intermediate quantities of low-level waste, between 9 and 37 percent of the maximum volume generated and within approximately 150 percent of the minimum volume, depending on the fuel group. For applicable fuel groups, conventional processing would produce the most low-level waste. In each case, continued wet storage would produce the next highest volume due to the combined effect of storage time and generation rate. When all applicable fuel groups are included, conventional processing would generate 138,200 cubic meters of low-level waste (29 percent of the SRS low-level waste 30-year forecast) and continued wet storage would generate 56,650 cubic meters (12 percent of the forecast). Of the two options being considered for the higher actinide targets, the Repackage and Prepare to Ship option would produce the smallest quantity of low-level waste, 32 percent of that estimated for Continued Wet Storage.

4.1.1.4.5 By-products of converting SNF into a waste form that is suitable for disposal in a geologic repository

With the exception of continued wet storage under the No-Action Alternative, the technology options would convert the fuels into a waste form that is likely to be suitable for permanent disposal in a geologic repository. The radioactive inventory in the final waste form would be substantially greater than 99 percent of the original fuel inventory. Very small amounts of residual radioactivity would remain in secondary low-level, hazardous/mixed low-

level, and transuranic waste streams as illustrated in Figures 4.1-1 through 4.1-7. SRS would use the surplus capacity in existing waste management facilities to treat, store, dispose of, or recycle the secondary waste in accordance with applicable regulations.

The melt and dilute and vitrification technologies would release from the fuel matrix volatile fission products (primarily cesium) from the fuel matrix which would be recovered as illustrated in Figure 4.1-3 and Figure 4.1-5. Residual cesium, strontium, and plutonium from conventional processing (as well as volatile fission products from melt and dilute, and vitrification technology options) would be moved from the high-level waste tanks and separated into a high volume – low radioactivity salt stream and a low volume – high radioactivity slurry. The salt stream would be approximately 95 percent of the total (before separation) volume and the slurry would capture approximately 99.999 percent of the cesium, strontium, and plutonium activity (Choi 1992). The slurry would be encapsulated in glass and poured into canisters at the Defense Waste Processing Facility. The canisters would be stored in a Glass Waste Storage Building for ultimate disposal in a geologic repository. The salt stream would be mixed into and solidified with concrete and disposed of in the Z-Area vaults.

4.1.1.4.6 Spent Fuel Canisters

DOE does not consider the SNF canisters resulting from alternate technology options to constitute a waste stream because they would be the end product of the new packaging options or new processing technology options being proposed. Nevertheless, the number of canisters is a useful measure of onsite storage space needed and the volume of the material that, after processing, could possibly be placed in a repository. Table 4.1-14 indicates the numbers of two types of canisters for the various technologies. The 17-inch canister would be used for co-disposal. The 24-inch canister would be used when the technology produces a vitrified product identical

| TC

| TC

| EC

EC |
TC |

Table 4.1-14. Numbers of spent fuel co-disposal and high-level waste canisters.

Technology	Co-Disposal or Direct Disposal canisters	24-inch high-level waste canisters
Prepare for direct co-disposal	1,400	NA ^a
Repackage and prepare to ship	NA ^b	1
Melt and dilute	400	10
Mechanical dilution ^c	630	10
Vitrification technologies ^d	1,350	10
Electrometallurgical treatment	—	90
Conventional processing ^e	—	150
Continued wet storage	—	41

- a. NA = not applicable, since DOE would use Co-Disposal.
b. Canisters would not be required to transfer material to another site.
c. Values were calculated for the press and dilute technology.
d. Values represent dissolve and vitrify and glass material oxidation and dissolution system technologies. The plasma arc technology would produce 490 canisters.
e. Values are for conventional processing the entire SNF inventory.

to the DPWF high-level waste borosilicate glass. After conventional processing, the 24-inch canisters would be stored in DWPF's Glass Waste Storage Building. The number of high-level waste canisters (Table 4.1-14) includes the secondary waste stream components generated by the technologies reported in Table 4.1-10.

4.1.1.5 Utility and Energy Resources

This section describes the estimated utility and energy requirements associated with each technology option under consideration in this EIS. Water, electricity, steam, and diesel fuel would be required to support many of the options. Estimates of water use include domestic water supplies and makeup water for process operations or equipment cooling. Steam is used primarily to heat facilities. Fuel consumption is based on use of diesel generators for backup power. Electrical requirements include that for normal office consumption such as heating, cooling, ventilation, and office equipment, and for specialized process-related equipment. The process equipment and the associated electrical demands would vary from option to option. All technologies would require canister loading and welding equipment. For the Melt and Dilute technology, the resistive heating associated with melting would require additional electricity. For aqueous processing, electrical requirements

would include the operation of canyon pumps, circulators or mixers, and denitrating equipment. For Vitrification, electrical equipment would be used for resistive heating and dissolution. For Electrometallurgical Treatment, electricity would be used for resistive melting of fuels, operation of an electrolytic bath for metal purification, final melting of the refined uranium product, and blending down with depleted uranium.

Tables 4.1-15 through 4.1-18 list estimated utility and energy requirements for the technology options applicable to each fuel group. For each option, this analysis considered three handling phases as potential sources of energy consumption: wet storage (pretreatment storage), treatment, and dry storage (post-treatment storage pending final disposition). The durations for these phases are provided in Appendix E. The period and utility use rate associated with each phase would vary depending on the fuel group and the option. As discussed above, DOE summed utility use from each phase; the values listed in the tables represent the total projected utility use for each option in a given fuel group.

DOE used annual utility consumption rates to calculate the estimates in the tables (Bickford et al. 1997); the rates are based on applicable cur-

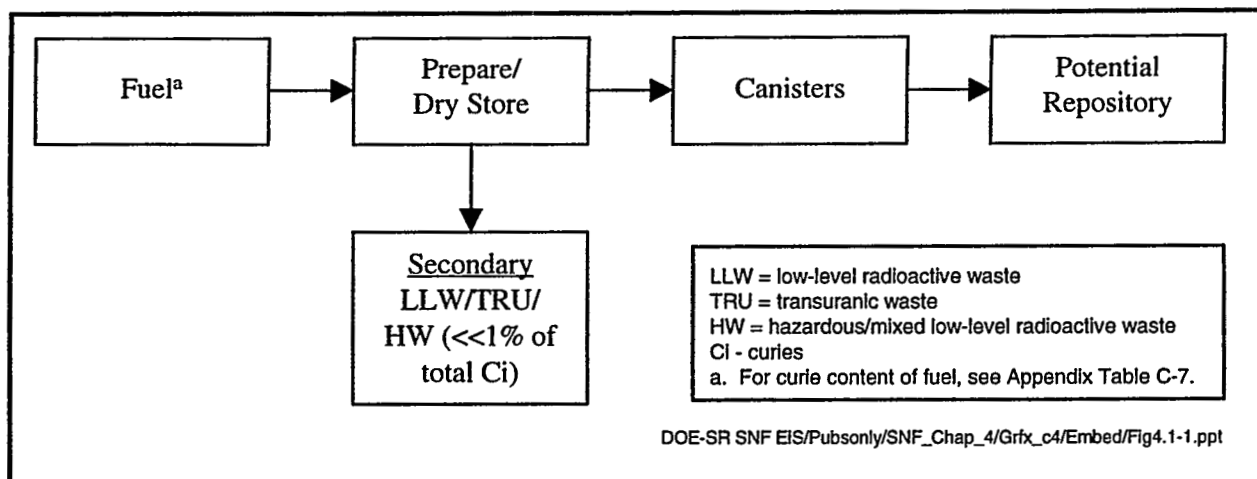


Figure 4.1-1. Type and source of waste streams generated by the Prepare for Direct Co-Disposal technology option.

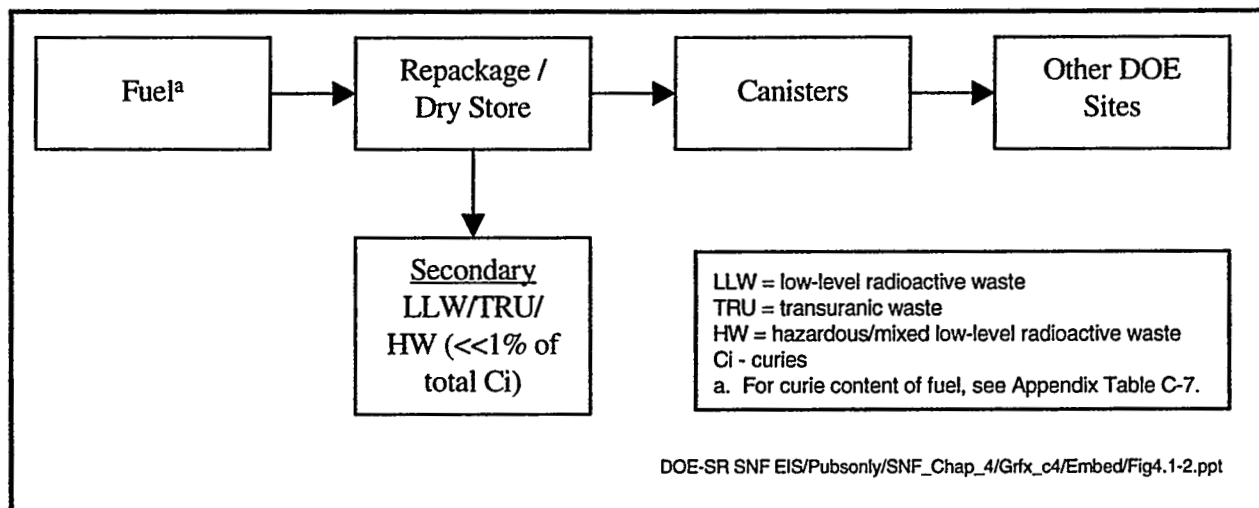


Figure 4.1-2. Type and source of waste streams generated by the Repackage and Prepare to Ship technology option.

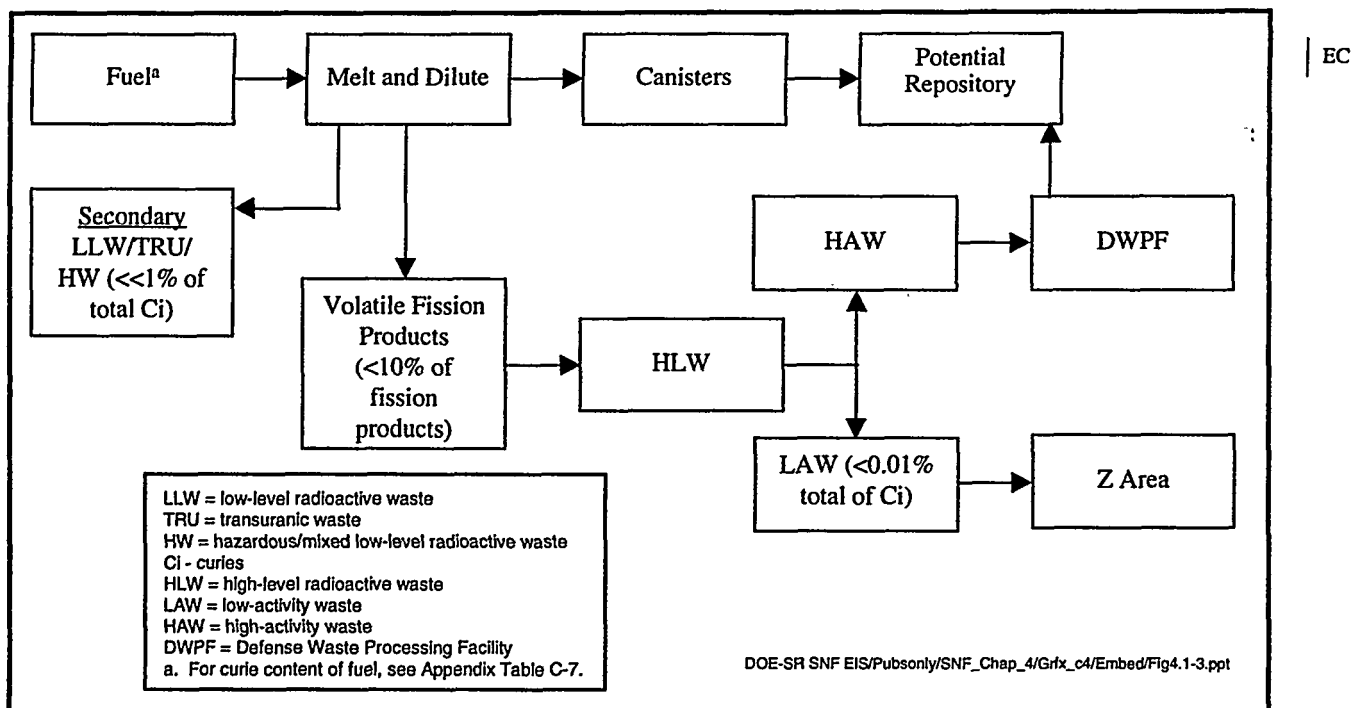


Figure 4.1-3. Type and source of waste streams generated by the Melt and Dilute technology option.

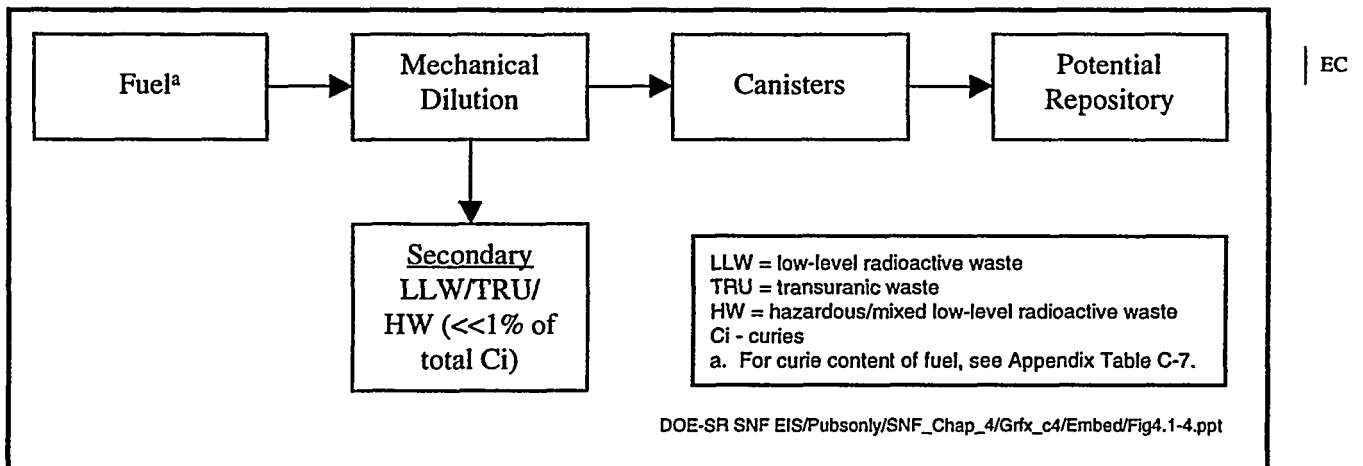


Figure 4.1-4. Type and source of waste streams generated by the Mechanical Dilution technology option.

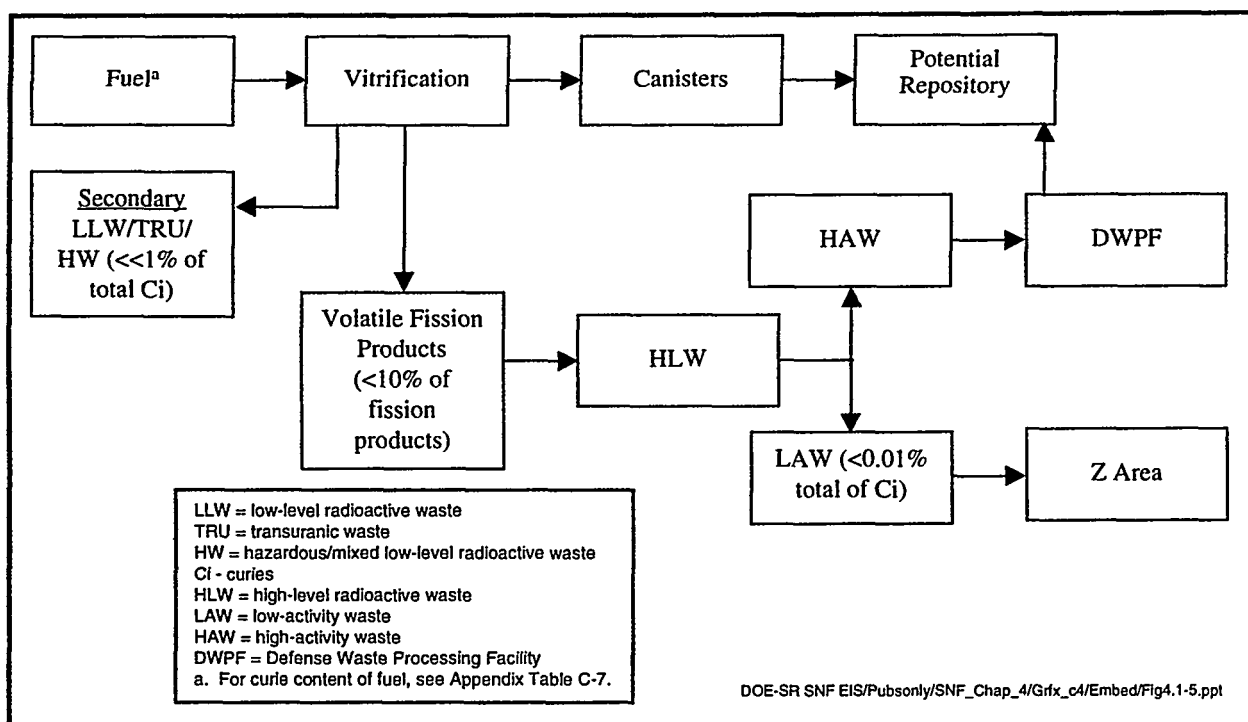


Figure 4.1-5. Type and source of waste streams generated by the Vitrification technology options.

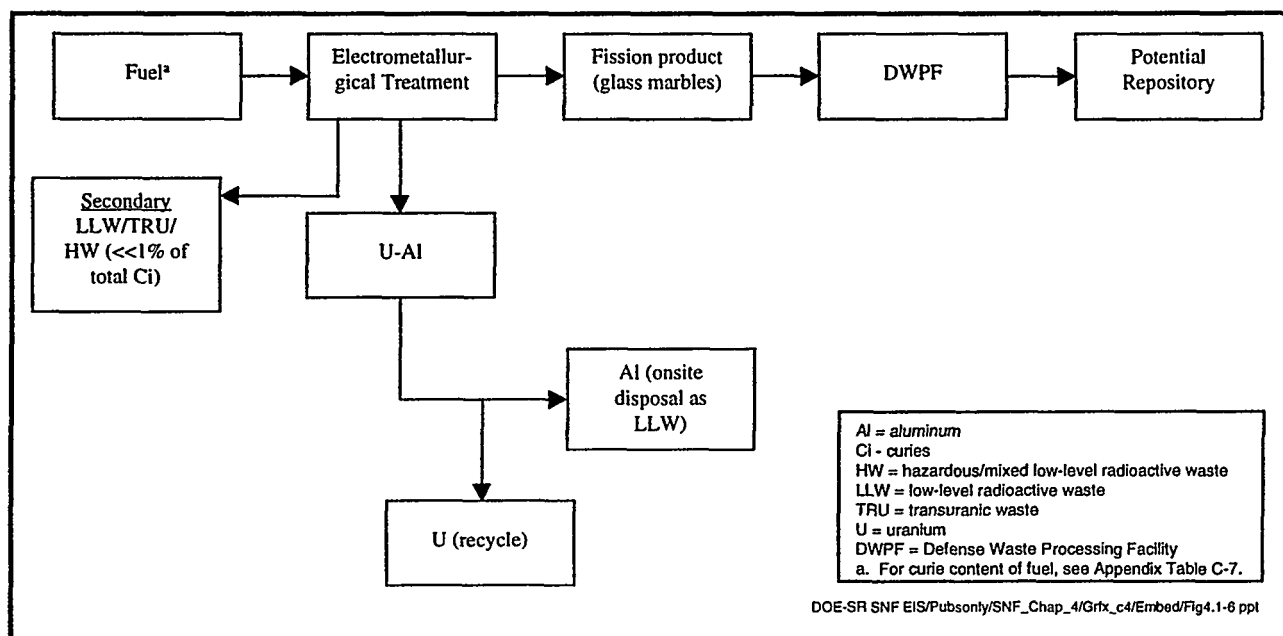
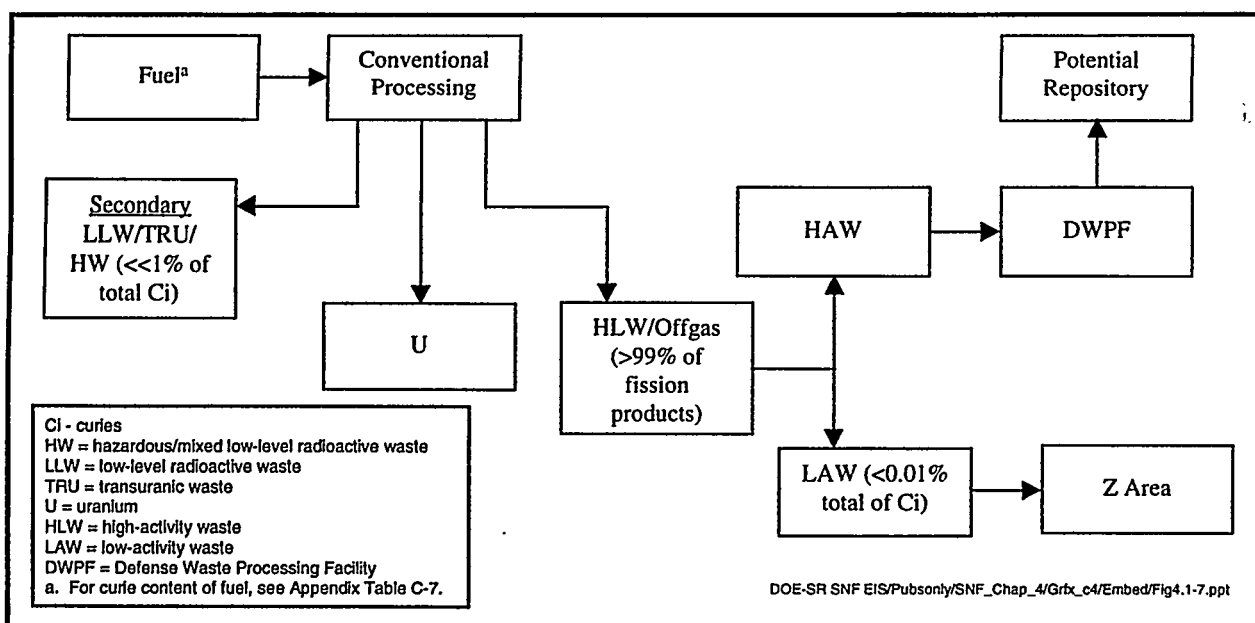


Figure 4.1-6. Type and source of waste streams generated by the Electrometallurgical Treatment technology option.



EC

Figure 4.1-7. Type and source of waste streams generated by the Conventional Processing technology option.

rent and past SRS operations or on engineering judgments for new treatment technologies.

The following paragraphs describe estimated utility requirements for the options.

4.1.1.5.1 Water Use

Vitrification and Electrometallurgical Treatment would require the most water, followed by Conventional Processing. Total requirements for Vitrification and Electrometallurgical Treatment of all applicable fuel groups would be less than 6,000 liters per year, (the equivalent of 4.3 gallons per day) which is a minute portion (0.00001 percent) of groundwater withdrawal of more than 5×10^9 liters per year (DOE 1997). Due to the comparatively long period required to process the HEU/LEU oxides and silicides requiring resizing or special packaging (Fuel Group C) and the loose uranium oxide in cans (Fuel Group D), the Conventional Processing technology would require the greatest amount of water for those groups. For the higher actinide targets, Repackage and Prepare to Ship would require 67 percent of the water needed to support the only other option under consideration for that fuel group, Continued Wet Storage. In

general, the Direct Disposal/Direct Co-Disposal, Melt and Dilute, Mechanical Dilution, and Repackage and Prepare to Ship technologies would require the least water for their applicable fuel groups, approximately 5 to 6 percent of the maximum requirement for a given group.

4.1.1.5.2 Electricity Use

Vitrification and Electrometallurgical Treatment would have the highest annual demand for electricity, followed by Conventional Processing. Differences in the time necessary to treat a fuel group under different options would affect total electricity requirements. Due to the longer period required to process the materials test reactor-like fuels (Fuel Group B), HEU/LEU oxides and silicides requiring resizing or special packaging (Fuel Group C), and loose uranium oxide in cans (Fuel Group D), Conventional Processing would require the most total electricity for those groups. For the higher actinide targets, Repackage and Prepare to Ship would require less than half the electricity needed to support continued wet storage. In general, for the appropriate fuel groups, the least electricity would be required to support Direct Co-Disposal and Mechanical Dilution.

Table 4.1-15. Water use (millions of liters) over the entire period of analysis.^a

Fuel group	Technologies							
	1	2	3	4	5	6	7	8
	Prepare for direct disposal/co-disposal	Repackage and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing	Continued wet storage
A. Uranium and Thorium Metal Fuels	9.7	NA	11	NA	170	170	66	17
B. Materials Test Reactor-Like Fuels	460	NA	520	440	7,900	7,900	4,800	790
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	130	NA	140	120	2,100	2,100	2,400	210
D. Loose Uranium Oxide in Cans	NA	NA	30	NA	440	440	690	45
E. Higher Actinide Targets	NA	4	NA	NA	NA	NA	NA	6
F. Non-Aluminum-Clad Fuels	NA	30	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.

HEU = Highly Enriched Uranium.

LEU = Low Enriched Uranium.

a. Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.

Table 4.1-16. Electricity use (megawatt-hours) over the entire period of analysis.^a

Fuel group	Technologies							
	1	2	3	4	5	6	7	8
	Prepare for direct disposal/co-disposal	Repackage and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing	Continued wet storage
A. Uranium and Thorium Metal Fuels	360	NA	1,300	NA	5,700	5,700	4,900	730
B. Materials Test Reactor-Like Fuels	17,000	NA	61,000	21,000	270,000	270,000	360,000	34,000
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	4,700	NA	16,000	5,500	72,000	72,000	180,000	9,000
D. Loose Uranium Oxide in Cans	NA	NA	3,400	NA	15,000	15,000	52,000	1,900
E. Higher Actinide Targets	NA	130	NA	NA	NA	NA	NA	270
F. Non-Aluminum-Clad Fuels	NA	1,100	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.

HEU = Highly Enriched Uranium.

LEU = Low Enriched Uranium.

a. Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.

Table 4.1-17. Steam use (millions of kilograms) over the entire period of analysis.^a

Fuel group	Technologies							
	1	2	3	4	5	6	7	8
	Prepare for direct disposal/co-disposal							
A. Uranium and Thorium Metal Fuels	3	NA	7	NA	20	20	29	6
B. Materials Test Reactor-Like Fuels	130	NA	330	130	940	940	2,200	250
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	35	NA	87	34	250	250	1,100	68
D. Loose Uranium Oxide in Cans	NA	NA	18	NA	52	52	310	14
E. Higher Actinide Targets	NA	1	NA	NA	NA	NA	NA	2
F. Non-Aluminum-Clad Fuels	NA	8	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.

HEU = Highly Enriched Uranium.

LEU = Low Enriched Uranium.

a. Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.

Table 4.1-18. Diesel fuel use (thousands of liters) over the entire period of analysis.^a

Fuel group	Technologies							
	1	2	3	4	5	6	7	8
	Prepare for direct disposal/co-disposal							
A. Uranium and Thorium Metal Fuels	2	NA	23	NA	1	1	180	4
B. Materials Test Reactor-Like Fuels	90	NA	1,100	80	45	45	13,000	170
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	24	NA	290	22	12	12	6,700	45
D. Loose Uranium Oxide in Cans	NA	NA	61	NA	3	3	2,000	10
E. Higher Actinide Targets	NA	1	NA	NA	NA	NA	NA	1
F. Non-Aluminum-Clad Fuels	NA	6	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.

HEU = Highly Enriched Uranium.

LEU = Low Enriched Uranium.

a. Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.

Annually, the maximum impact alternative electrical demand is 23,600 megawatt-hours, which is approximately 3.5 percent of the current SRS annual usage of 660,000 megawatt-hours.

4.1.1.5.3 Steam Use

Where applicable, Conventional Processing would have the highest annual demand for steam. For higher actinide targets, Repackage and Prepare to Ship would require half the steam needed to support continued wet storage. In general, Direct Co-Disposal and Mechanical Dilution would require the least steam.

4.1.1.5.4 Diesel Fuel Use

For several options, DOE would use diesel fuel to support SNF treatment and storage. On an annual basis, Conventional Processing and Melt and Dilute would need the most diesel fuel. The least diesel fuel would be associated with the Vitrification and Electrometallurgical Treatment technologies, because both would require fuel only to support initial wet storage. The two options that DOE is considering for the higher actinide targets (Repackage and Prepare to Ship and Continued Wet Storage) would require comparable amounts of diesel fuel.

4.1.1.6 Environmental Justice

This section examines whether minority or low-income communities (as defined in Section 3.5.3) could receive disproportionately high and adverse human health and environmental impacts as a result of the actions described in this EIS. Even though DOE does not anticipate adverse health impacts from the options, it analyzed for the possibility of "disproportionately high and adverse human health or environmental effects on minority populations or low-income populations" (Executive Order 12898). Figures 3.5-1 and 3.5-2 show minority and low-income communities by census tract. This section discusses average radiation doses that individuals in those communities could receive and compares them to predicted doses that individu-

als in the other communities within the 80-kilometer- (50-mile) radius region could receive.

Figure 4.1-8 has SRS as the center of a circle with 22.5-degree sectors and concentric rings from 10 to 50 miles (16 to 80 kilometers) out from the center at 10-mile (16-kilometer) intervals. For this analysis, DOE calculated a fraction of the total population dose for each sector, laid the sector circle over the census tract map, and assigned each tract to a sector. If a tract fell in more than one sector, DOE assigned it to the sector with the largest dose value.

DOE analyzed impacts by comparing the per capita dose that each type of community would receive to doses other types of communities in the same ring would receive. To eliminate the possibility of diluting and masking impacts to a low-population community close to SRS with a high dose per person by including them with impacts to a high-population community farther from the Site, the analysis made comparisons in a series of concentric circles, the radii of which increase in 10-mile (16-kilometer) increments.

To determine the radiation dose received per person in each type of community, the analysis multiplied the number of people in each tract by that tract's dose value to obtain a total community population dose for each tract, summed these population doses in each concentric circle, and divided by the total community population in the circle to get a community per capita dose for each area of the circle. Because the per capita dose for communities (Table 4.1-19) would be constant for every alternative, the relative differences in impacts between communities would also be constant. Thus, Figure 4.1-9 and Table 4.1-19 indicate the distribution of per capita doses to types of communities in the 50-mile (80-kilometer) region. As shown in Figure 4.1-9, atmospheric releases would not disproportionately affect minority communities (population equal to or greater than 35 percent of the total population) or low income (equal to or greater than 25 percent of the total population) in the 50-mile region; that is, a comparison

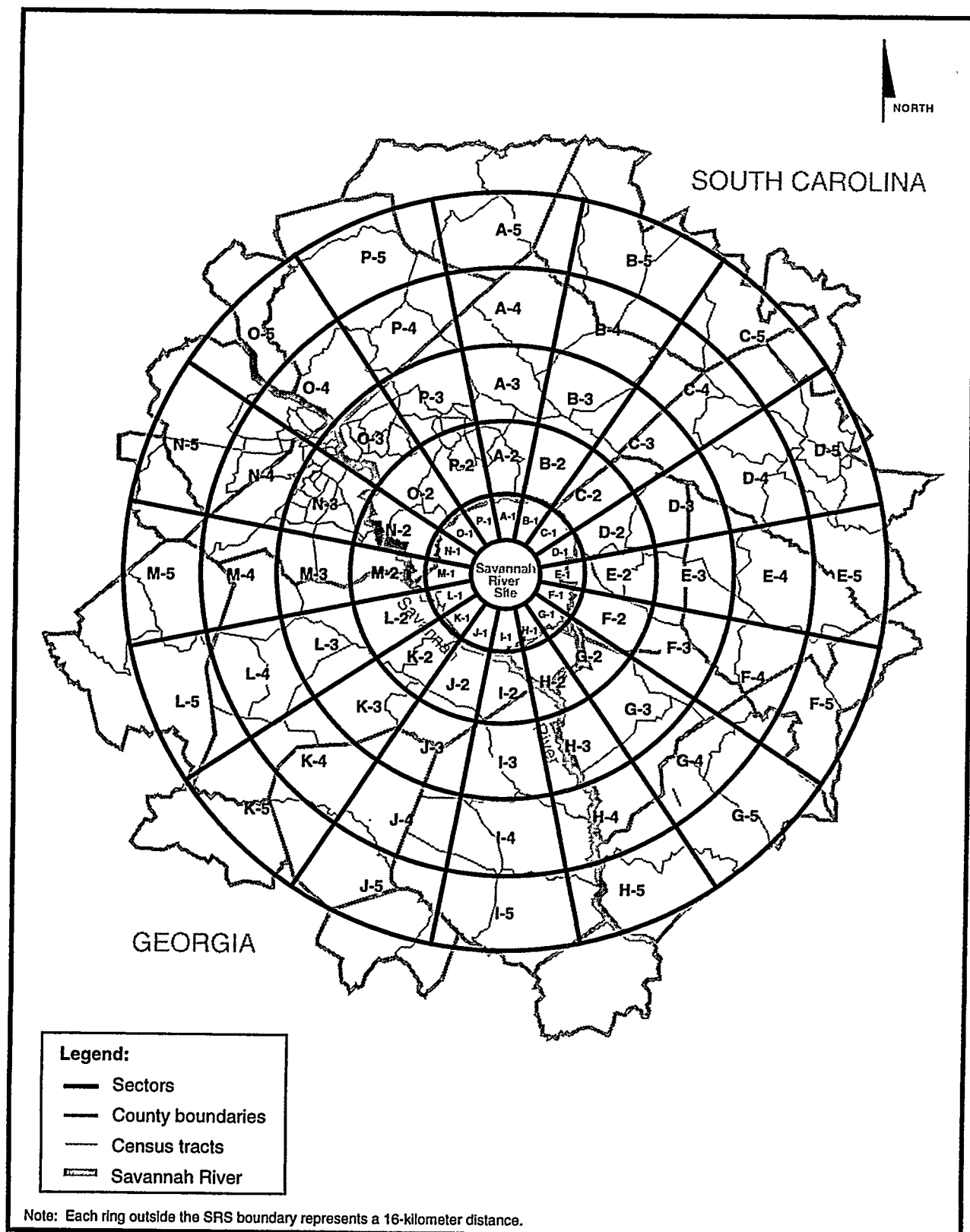


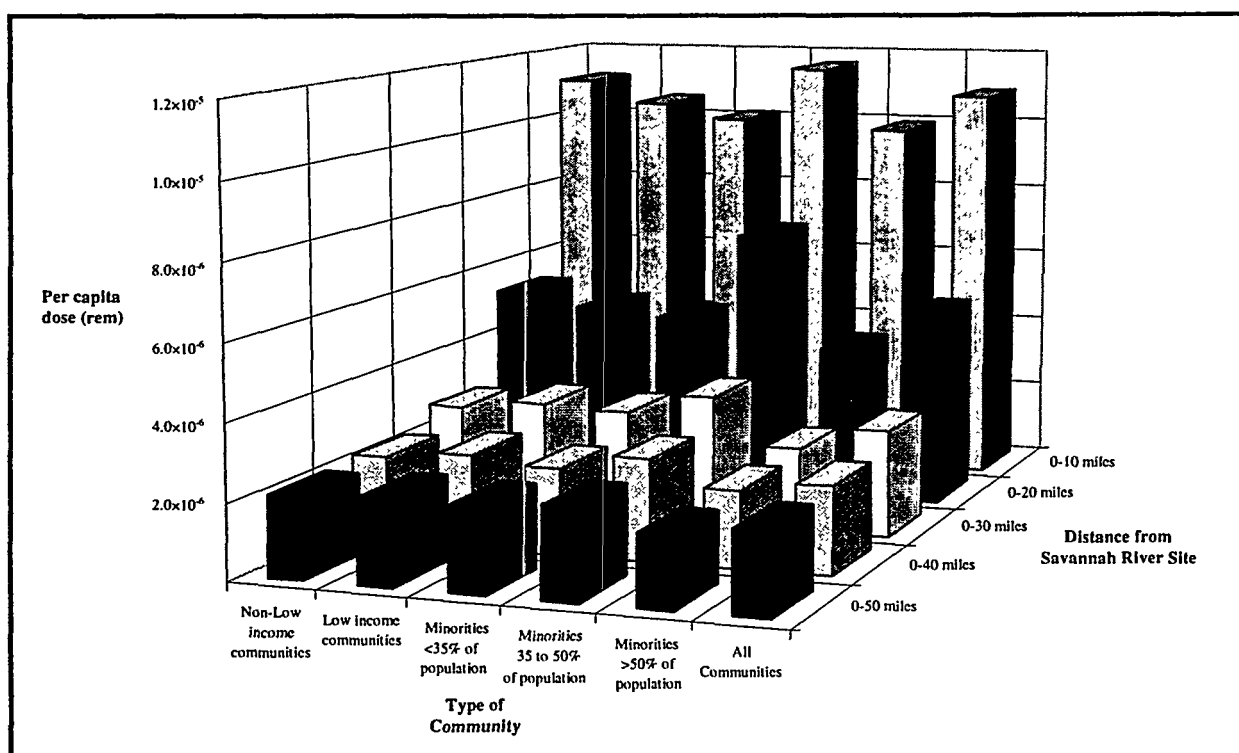
Figure 4.1-8. Annular sectors around the Savannah River Site.

NW SNF EIS/PubsOnly/SNF Chap_4/Grfx_c4/14-1-8.ai

Table 4.1-19. Estimated per capita annual dose (rem) for identified communities in 80-kilometer (50-mile) region.^a

Distance	Low income		Minorities			All communities (rem)
	Less than 25 percent of population (rem)	Equal to or more than 25 percent of population (rem)	Less than 35 percent of population (rem)	35 percent to 50 percent of population (rem)	Equal to or more than 50 percent of population (rem)	
0-10 miles (0-16 km ^b)	1.1×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.2×10^{-5}	1.0×10^{-5}	1.1×10^{-5}
0-20 miles (0-32 km)	5.0×10^{-6}	5.0×10^{-6}	5.0×10^{-6}	7.0×10^{-6}	4.0×10^{-6}	5.0×10^{-6}
0-30 miles (0-48 km)	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	2.0×10^{-6}	3.0×10^{-6}
0-40 miles (0-64 km)	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	3.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}
0-50 miles (0-80 km)	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}

- a. Per capita dose based on a population dose of 1 person-rem. Per capita doses for other population doses can be obtained by multiplying the values in this table by the population dose.
- b. km = kilometers.

**Figure 4.1-9.** Distribution of a hypothetical unit population dose among SRS communities.

of per capita doses indicates that they do not vary greatly.

For example, DOE used an annual total population dose of 1 person-rem to prepare Figure 4.1-9 and its supporting data in Table 4.1-19. In comparison, the maximum annual total population dose of 0.56 person-rem for the maximum impact alternative (see Section 4.1.2) would result in 56 percent of the impact shown in Figure 4.1-9 and Table 4.1-19. For any other population dose, the per capita dose for communities can be determined by multiplying that population dose by the values listed in Table 4.1-19.

The distribution of carcinogenic and criteria pollutant emissions from routine operations and of criteria pollutants from construction activities would be essentially identical to those described for airborne radiological emissions because the distribution pathways would be the same. As a result, nonradiological emissions from any option would not cause disproportionate impacts on minority or low-income communities. Because nonradiological pollutant emissions would cause minimal impacts for any option, and because there would not be disproportionate distribution of these impacts among types of communities, environmental justice concerns would not be associated with the alternatives.

4.1.1.7 Transportation

This section discusses the potential radiological consequences of the onsite transportation of SNF and the potential consequences of transportation to a geologic repository. All onsite shipments (those that originate and terminate on SRS) would be by rail. Movements of SNF within an SRS area (e.g., H Area or F Area) are operational transfers, not onsite shipments. The potential consequences of shipping SNF from the SRS to a geologic repository are a conservative (based on worst-case number of shipments and mode of transportation) representation of impacts based on preliminary information. The full analysis of transportation impacts will be included in the EIS for a Geological Repository for the Disposal of Spent Nuclear Fuel and

High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada (currently in preparation).

4.1.1.7.1 Onsite Incident-Free Transportation Analysis [SRS]

The analysis assumed a crew of four engineers for each shipment and that the external dose rate 6.6 feet (2 meters) from the shipping cask was 100 millirem per hour (HNUS 1994a), which is the SRS procedurally-allowed maximum dose rate during onsite fuel shipments. Actual receptor dose rates would depend on receptor distance from the shipping cask (39.4 feet [12 meters]). The duration of exposure would depend on the transport vehicle speed. In addition, vehicle crew time would depend on the distance of each shipment.

Table 4.1-20 summarizes the collective doses (person-rem) and health effects (latent cancer fatalities) associated with a single incident-free onsite shipment of SNF at SRS.

To determine the incident-free transportation dose for management of all SRS spent nuclear fuel, it is necessary to calculate the total dose over all shipments. DOE has estimated that it would take approximately 150 rail shipments to de-inventory the Receiving Basin for Offsite Fuels to the L-Area Disassembly Basin. This action would occur under all alternatives, including the No-Action Alternative. The radiation dose to the crew from these shipments is estimated to be approximately 0.57 person-rem, which could result in 2.3×10^{-4} latent cancer fatalities.

DOE has estimated that it would take approximately 300 rail shipments to transport the contents of the L-Area Disassembly Basin (including the fuel that was previously in the Receiving Basin for Offsite Fuels) to the Transfer and Storage Facility; the Transfer, Storage, and Treatment Facility; or the F- and H-Area Canyons. This action would occur under all alternatives, except the No-Action Alternative. Assuming the bounding location for the

TC

TC

TC

TC

TC

Table 4.1-20. Collective doses and health effects for onsite incident-free SNF shipments.^a

Shipment origin/destination	Crew dose per shipment (person-rem)	Number of LCFs ^b per shipment
		Crew
L Area/H Area	3.80×10^{-3}	1.52×10^{-6}
L Area/F Area	4.10×10^{-3}	1.64×10^{-6}
F Area/H Area	1.40×10^{-3}	5.60×10^{-7}
P Area/H Area	4.90×10^{-3}	1.96×10^{-6}
P Area/F Area	3.88×10^{-3}	1.55×10^{-6}
C Area/H Area	3.33×10^{-3}	1.33×10^{-6}
C Area/F Area	4.20×10^{-3}	1.68×10^{-6}

a. Derived from HNUS (1994a).

b. LCF = latent cancer fatality.

Transfer and Storage Facility or the Transfer, Storage, and Treatment Facility, the radiation dose to the crew from these shipments is estimated to be approximately 1.23 person-rem which could result in 4.9×10^{-4} latent cancer fatalities. Therefore, for the No-Action Alternative, the total radiation dose to the shipping crew would be approximately 0.57 person-rem, which could result in 2.3×10^{-4} latent cancer fatalities. For all other alternatives, the total radiation dose to the crew would be approximately 1.8 person-rem, which could result in 7.2×10^{-4} latent cancer fatalities.

4.1.1.7.2 Incident-Free Transportation Analysis [Geologic Repository]

DOE estimated the impacts of shipping SNF from SRS to a theoretical geologic repository in the Western United States (approximately 4,000 kilometers [2,500 miles] from SRS) by truck. This analysis assumes all shipments from SRS, approximately 1,400 (worst case among the alternatives), would be by truck because the impacts would bound the impacts of rail shipments. Because the transport of SRS spent fuel would use existing highways, it would represent a very small fraction of national highway traffic. Consequently, there would be negligible impacts on land use; air quality; hydrology; biological resources and cultural resources; socioeconomic; noise; aesthetics; utilities, energy, and materials; or waste management. The analysis of the potential impacts of transporting

SRS spent nuclear fuel to the repository focuses on the potential radiological impacts to workers and the public.

DOE recognizes that it cannot predict with any certainty the specific routes that would be used to ship SNF to a repository. Nonetheless, the analysis uses current regulations governing highway shipments to select actual highway routes to estimate the potential environmental impacts of national transportation. Assumed distances within the various rural, suburban, and urban population zones can be found on Table 4.1-21.

Loading Operations

Prior to shipping the fuel, DOE would load it into NRC certified Type B shipping casks. The potential dose to involved workers from the loading operation would be less than that expected at a commercial nuclear facility because the radionuclide inventory of commercial fuel is higher than that of the DOE SNF. The dose would be further limited by worker rotation and other administrative controls. DOE expects any dose to uninvolved workers would be negligible because they would not have tasks that could result in radiation exposure. Likewise, DOE expects radiation exposure to the public would not occur because of the distance of the loading operations from the areas of public access.

Table 4.1-21. Incident-free radiological impacts of 1,400 offsite truck shipments of spent nuclear fuel to the proposed Yucca Mountain Geologic Repository.

Exposure group	Unit risk factors (person-rem kilometer) ^a			Kilometers traveled		
	Rural	Suburban	Urban	Rural	Suburban	Urban
Occupational	4.6×10 ⁻⁵	1.0×10 ⁻⁴	1.7×10 ⁻⁴	3,292.6	570.2	65.9
Off-link ^b	1.2×10 ⁻⁷	1.6×10 ⁻⁵	1.1×10 ⁻⁴	3,292.6	570.2	65.9
On-link ^c	5.0×10 ⁻⁶	1.5×10 ⁻⁵	1.5×10 ⁻⁴	3,292.6	570.2	65.9
Stops	1.2×10 ⁻⁴	1.2×10 ⁻⁴	1.2×10 ⁻⁴	3,292.6	570.2	65.9
	Collective dose (person-rem)			Total collective dose	LCF ^d	
	Rural	Suburban	Urban			
Occupational	212	80	16	308		0.123
General population						
Off-link ^b	1	13	10	24		0.012
On-link ^c	23	12	14	49		0.024
Stops	553	96	11	660		0.330
General population total						0.366

a. The methodology, equations, and data used to develop the unit risk factors are discussed in Madsen et al. (1986) and Neuhauser and Kanipe (1992). Cashwell et al. (1986) contains a detailed explanation of the use of unit risk factors.

b. Off-link general population are persons within 800 meters (2,625 feet) of the highway.

c. On-link general population are persons sharing the highway.

d. LCF = latent cancer fatality.

Transportation to a Geologic Repository

To estimate the potential impacts of incident-free transportation of SNF to a repository, the analysis considered both the public and workers. Unit risk factors commonly used in a number of other DOE EISs were used to determine the potential person-rem exposure per kilometer for both workers and public. In the case of the general population, both off-link and on-link doses were calculated. The off-link dose could affect persons within 800 meters (2,625 feet) of the highway; the on-link dose could affect persons sharing the highway. Table 4.1-21 presents the potential incident-free radiological impacts from 1,400 shipments of SNF from the SRS to a theoretical geologic repository. As can be seen from the table, potential latent cancer fatalities could result in less than 1 additional death from radiation over the life of the shipments.

4.1.1.7.3 Onsite Transportation Accident Analysis [SRS]

DOE analyzed radiological impacts from potential accidents to the onsite maximally exposed individual from onsite rail shipments. The analysis calculated doses using the RADTRAN computer code (Neuhauser and Kanipe 1992) with site-specific meteorology, and calculated risk using site-specific rail accident rates and accident probabilities (HNUS 1994b).

The analysis assumed a release of the maximum reasonably foreseeable amount of radioactive material for the type of SNF shipped on SRS (HNUS 1994b). Radiological doses were modeled for three human receptor groups: the onsite worker population, members of the public residing near SRS, and the maximally exposed offsite individual. The consequences are expressed as excess latent cancer fatalities in each receptor group.

Table 4.1-22 summarizes the radiation doses resulting from the most severe reasonably foreseeable onsite transportation accident and associated latent cancer fatalities.

4.1.1.7.4 Transportation Accident Analysis [Geologic Repository]

Potential impacts from accidents resulting from transporting SNF to a geologic repository are not quantified in this document but have been analyzed in the EIS for a Geologic Repository for Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada. Previous EISs, including the Foreign Research Reactor Spent Fuel EIS (DOE 1996) and the Programmatic Spent Fuel EIS (DOE 1995b) analyzed the potential accident impacts of transporting SNF. The following discussions summarize the types of accidents that could be expected. Impacts are presented in Table 4.1-23.

Loading Operation

In general, accidents from loading operations could be caused by unplanned contact (bumping) during lifting or handling of casks, canisters, or fuel assemblies. Initiating events could include fires, explosions, earthquakes, cask tor

nadoes, canister or basket drops, and loaded shipping drops. The Interim Management of Nuclear Materials at SRS EIS (DOE 1995a) assessed the radiological impacts from potential accidents associated with preparing, storing, and onsite shipment of some spent nuclear fuel.

Transportation to a Geologic Repository

Several types of accidents potentially could occur while transporting SNF. The first type of accident, resulting in the most radiological exposure to the public, assumes the breach of a shipping cask during an accident resulting in the release of a fraction of its contents to the air. This accident would be very unlikely. The second type of accident would involve truck wrecks that could result in non-radiological fatalities to workers or members of the public. The probability of an accident is dependent upon the number of shipments made and total miles traveled.

4.1.2 IMPACTS OF THE ALTERNATIVES

As discussed in Chapter 2, none of the options for the management of SNF, except Continued Wet Storage, would address the requirements of all six fuel types. Therefore, DOE must consider combinations of technologies to satisfy the purpose and need identified in Chapter 1. This

Table 4.1-22. Impacts on SRS workers, maximally exposed offsite individuals, and offsite population from SNF transportation accidents on Savannah River Site.

Accident frequency	Worker dose (rem)	Probability of a worker LCF ^b	MEI ^c dose (rem)	Probability of a LCF to the MEI	Population dose (person-rem)	Population LCFs
1.28×10^{-4}	2.78	1.11×10^{-3}	2.2×10^{-5}	1.08×10^{-8}	0.16	8.21×10^{-5}

a. Source: DOE (1995a).

b. LCF = latent cancer fatality.

c. MEI = maximally exposed individual.

Table 4.1-23. Truck transportation accident analysis impacts.

Radiological impacts				Traffic impacts		
Risk factor (person-rem/shipment) ^a	Maximum number shipments	Total (person-rem)	Total LCFs	Risk factor (fatality/shipment) ^b	Maximum number shipments	Total fatality
1.79×10^{-5}	1,400	0.025	1.25×10^{-5}	1.12×10^{-4}	1,400	0.16

LCF = latent cancer fatalities.

a. DOE (1996).

b. Adapted from DOE (1999).

section provides the results of analyzing combinations of the technology options applicable to the fuel groups. Excluding continued wet storage, there are more than 700 combinations of technology options and fuel groups that could be analyzed. However, it would be impractical and unreasonable to do so. DOE has identified four sets of combinations for analysis as alternatives in this EIS (in addition to No Action) which it believes are representative. These four alternatives are the Minimum Impact Alternative, Direct Disposal Alternative, Preferred Alternative, and Maximum Impact Alternative. The data in Section 4.1.1 can be used to compile the impacts of other configurations of viable cases.

Continued wet storage for all fuel types is the No-Action Alternative. National Environmental Policy Act (NEPA) regulations require the evaluation of No Action, (which would not meet the purpose and need described in Chapter 1); however, it provides a baseline against which DOE can compare the action alternative combinations.

The second alternative, Minimum Impact, would result in the smallest environmental impacts to human health. It is also the environmentally-preferred alternative.

The third alternative is Direct Disposal. All fuel types that could be dry stored would be. Higher Actinide Targets and Non-Aluminum-Clad Fuels would be Repackaged and Prepared to Ship Offsite. Uranium and Thorium Metal Fuels and Loose Uranium Oxide in Cans would undergo conventional processing.

The fourth alternative is the Preferred Alternative. Melt and Dilute would be used to treat the Materials Test Reactor-like fuels, most of the HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging (Group C), and most of the Loose Uranium Oxide in Cans (Group D). Group A and the remaining Group C and Group D fuels (<10 percent of the material in these fuel groups) would be treated

with conventional processing. Finally, the Higher Actinide Targets and the Non-Aluminum-Clad fuels would be Repackaged and Prepared to Ship offsite.

The final alternative would apply the chemical processing option to all the fuel except the higher actinide targets and non-aluminum-clad SNF and probably would produce the greatest environmental impacts, and therefore, provides an upper bound. It is termed the Maximum Impact Alternative. Section 2.4 provides a complete description of the SNF management alternatives.

Tables 4.1-24 through 4.1-26 list the impacts of the five alternatives summed from the operational impacts of each appropriate technology presented in Section 4.1.1. The following sections describe the alternatives and the bases for their selection. The conclusions from Section 4.1.1.5 on environmental justice would apply to all the alternatives.

DOE based the values listed for annual radiation dose to the noninvolved worker, the offsite maximally exposed individual, and the 620,000-person population surrounding SRS on the sum of the annual doses for each technology-fuel group included in the alternative. Since the time intervals over which these annual doses would occur might not coincide, this method could overestimate the annual doses that actually would occur.

The values in Table 4.1-26 for health effects to the noninvolved worker, maximally exposed individual, and the offsite population for the No-Action Alternative represent current reactor area emissions (including two SNF wet basins) for the entire period of analysis. The values for the other alternatives would be incremental above these baseline values. Summing these baseline and incremental values would be conservative, however, because there would not be two SNF wet basins operating over the entire 38-year period of analysis.

Table 4.1-24. Estimated maximum incremental concentrations of nonradiological air pollutants for the noninvolved worker.

Pollutant	Averaging Time	Regulatory Standard ^a	No Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative
Toxic Pollutants (mg/m³)							
Nitric acid	24-hour	5	0.03	0.02	2.75	2.62	7.95
1,1,1-Trichloroethane	24-hour	1,900	–	–	0.02	0.02	0.05
Benzene	24-hour	3.19	–	–	0.02	0.02	0.05
Ethanolamine	24-hour	6	0.03	0.02	0.02	0.02	0.03
Ethyl benzene	24-hour	435	–	–	0.01	0.01	0.02
Ethylene glycol	24-hour	None	0.03	0.02	0.02	0.02	0.03
Formaldehyde	24-hour	0.75	0.03	0.02	0.02	0.02	0.03
Glycol ethers	24-hour	80	0.03	0.02	0.02	0.02	0.03
Hexachloronaphthalene	24-hour	0.2	0.03	0.02	0.02	0.02	0.03
Hexane	24-hour	1,800	0.03	0.02	0.03	0.03	0.06
Manganese	24-hour	5	–	–	0.01	0.01	0.02
Mercury	24-hour	0.1	–	–	0.01	0.01	0.02
Methyl alcohol	24-hour	260	0.03	0.02	0.02	0.02	0.03
Methyl ethyl ketone	24-hour	590	0.03	0.02	0.02	0.02	0.03
Methyl isobutyl ketone	24-hour	410	–	–	0.01	0.01	0.02
Methylene chloride	24-hour	86.7	–	–	0.02	0.02	0.05
Napthalene	24-hour	50	0.03	0.02	0.02	0.02	0.03
Phenol	24-hour	19	–	–	0.01	0.01	0.02
Phosphorus	24-hour	0.1	–	–	0.01	0.01	0.02
Sodium hydroxide	24-hour	2.0	–	–	0.01	0.01	0.02
Toluene	24-hour	754	0.03	0.02	0.03	0.03	0.06
Trichloroethene	24-hour	537	–	–	0.01	0.01	0.02
Vinyl acetate	24-hour	None	–	–	0.01	0.01	0.02
Xylene	24-hour	435	0.03	0.02	0.05	0.05	0.10
Criteria Pollutants (µg/m³)							
Nitrogen oxides	Annual	NA	–	0.05	38.2	36.4	111
Total Suspended Particulates (total dust)	8-hour	15	–	0.02	0.35	0.34	0.99
Particulate Matter (<10 µm)	8-hour	5	–	0.09	0.08	0.08	0.05
	24-hour	NA	–	0.99	0.86	0.87	0.62
Carbon monoxide	8-hour	55	0.03	0.25	1.81	1.82	4.78
	1-hour	NA	0.03	0.79	5.65	5.68	14.93
Sulfur dioxide	Annual	NA	–	0.02	0.04	0.04	0.08
	8-hour	13	–	0.02	0.31	0.30	0.86
	3-hour	NA	–	0.02	0.72	0.70	2.07
Gaseous fluorides	1-month	None	–	–	0.10	0.10	0.29
	1-week	NA	–	–	0.18	0.17	0.52
	24-hour	NA	–	–	0.55	0.52	1.59
	12-hour	NA	–	–	0.80	0.76	2.32
Ozone (as VOC)	1-hour	0.2	–	nc	nc	nc	nc

– = no air emission associated with this combination.

NA = not applicable.

nc = not calculated.

VOC = volatile organic compound.

a. 29 CFR 1910.1000, Subpart Z and OSHA 8-hour time-weighted averages.

Table 4.1-25. Estimated maximum incremental concentrations of nonradiological air pollutants at the Site boundary.

Pollutant	Averaging Time	Regulatory Standard ^a	No Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative	
Toxic Pollutants (mg/m³)								
Nitric acid	24-hour	125	—	—	0.11	0.10	0.31	
1,1,1-Trichloroethane	24-hour	9,550	0.03	0.03	0.03	0.03	0.03	
Benzene	24-hour	150	—	—	0.01	0.01	0.02	
Ethanolamine	24-hour	200	0.03	0.03	0.03	0.03	0.03	
Ethyl benzene	24-hour	4,350	—	—	0.01	0.01	0.02	
Ethylene glycol	24-hour	650	0.03	0.03	0.03	0.03	0.03	
Formaldehyde	24-hour	15	0.03	0.03	0.03	0.03	0.03	
Glycol ethers	24-hour	+	0.03	0.03	0.03	0.03	0.03	
Hexachloronaphthalene	24-hour	1	0.03	0.03	0.03	0.03	0.03	
Hexane	24-hour	200	0.03	0.03	0.03	0.03	0.03	TC
Manganese	24-hour	25	—	—	0.01	0.01	0.02	
Mercury	24-hour	0.25	—	—	0.01	0.01	0.02	
Methyl alcohol	24-hour	1,310	0.03	0.03	0.03	0.03	0.03	
Methyl ethyl ketone	24-hour	14,750	0.03	0.03	0.03	0.03	0.03	
Methyl isobutyl ketone	24-hour	2,050	—	—	0.01	0.01	0.02	
Methylene chloride	24-hour	8,750	—	—	0.01	0.01	0.02	
Napthalene	24-hour	1,250	0.03	0.03	0.03	0.03	0.03	
Phenol	24-hour	190	—	—	0.01	0.01	0.02	
Phosphorus	24-hour	0.5	—	—	0.01	0.01	0.02	
Sodium hydroxide	24-hour	20	—	—	0.01	0.01	0.02	TC
Toluene	24-hour	2,000	0.03	0.03	0.03	0.03	0.03	
Trichloroethene	24-hour	6,750	—	—	0.01	0.01	0.02	
Vinyl acetate	24-hour	176	—	—	0.01	0.01	0.02	
Xylene	24-hour	4,350	0.03	0.03	0.03	0.03	0.03	
Criteria Pollutants (µg/m³)								
Nitrogen oxide	Annual	100	0.03	0.02	1.17	1.12	3.36	
Total Suspended Particulates	Annual	75	0.03	0.02	0.02	0.02	0.02	
Particulate Matter (<10 µm)	Annual	50	—	—	0.01	0.01	0.02	
	24-hour	150	—	—	0.05	0.04	0.13	
Carbon monoxide	8-hours	10,000	0.03	0.07	0.49	0.50	1.31	
	1-hour	40,000	0.03	0.37	3.60	3.57	9.76	
Sulfur dioxide	Annual	80	—	0.02	0.02	0.02	0.02	
	24-hour	365	—	0.03	0.07	0.07	0.13	
	3-hour	1300	—	—	0.34	0.32	0.98	
Gaseous fluoride	1-month	0.8	—	—	0.01	0.01	0.02	
	1-week	1.6	—	—	0.02	0.01	0.04	
	24-hour	2.9	—	—	0.03	0.02	0.07	
	12-hour	3.7	—	—	0.05	0.04	0.13	
Ozone (as VOC)	1-hour	235	—	0.16	0.38	0.41	0.80	

— = no air emission associated with this option.

+ = no state standard.

VOC = volatile organic compound.

a. SCDHEC standard No. 2 (criteria pollutants) and No. 8 (toxic pollutants).

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Table 4.1-26. Impacts from alternatives.^a

Impact	No Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative ^b	Maximum Impact Alternative
Health Effects for the Entire Period of Analysis (1998-2035)^c					
MEI ^e dose (millirem)	0.63 ^d	6.1×10 ⁻⁴	7.2×10 ⁻³	0.19	0.67
MEI LCF ^e probability	3.1×10 ^{-7d}	3.0×10 ⁻¹⁰	3.6×10 ⁻⁹	9.5×10 ⁻⁸	3.4×10 ⁻⁷
Population dose (person-rem)	22.6 ^d	0.022	0.077	6.9	8.7
Population LCFs (unitless)	0.011 ^d	1.1×10 ⁻⁵	3.8×10 ⁻⁵	3.4×10 ⁻³	4.4×10 ⁻³
Collective worker dose (person-rem)	760	690	840	841	2,100
Collective worker LCFs (unitless)	0.30	0.28	0.34	0.33	0.84
Noninvolved worker dose (millirem)	4.25 ^d	5.0×10 ⁻³	0.02	1.53	1.53
Noninvolved worker LCF probability	1.7×10 ^{-6d}	2.0×10 ⁻⁹	9.6×10 ⁻⁹	6.1×10 ⁻⁷	6.3×10 ⁻⁷
Annual Radiological Air Emission Impacts					
Maximum annual MEI ^d dose (millirem)	0.02 ^d	6.1×10 ⁻⁴	7.4×10 ⁻⁴	0.044	0.015
Maximum annual population dose (person-rem)	0.59 ^d	0.022	0.027	1.6	0.56
Maximum annual noninvolved worker dose (millirem)	0.11 ^d	5.0×10 ⁻³	6.0×10 ⁻³	0.36	0.12
Annual Radiological Liquid Emission Impacts					
Maximum annual MEI dose (millirem)	0	0	1.4×10 ⁻³	4.2×10 ⁻⁵	0.057
Maximum annual population dose (person-rem)	0	0	4.9×10 ⁻³	2.4×10 ⁻⁴	0.19
Waste Generation (cubic meters) for the Entire Period of Analysis (1998-2035)					
High-level waste					
Liquid	2,300	660	1,200	1,050	10,500
Equivalent DWPF canisters	38	11	20	17	160
Saltstone	6,100	1,800	3,200	2,700	27,000
Transuranic waste	0	15	360	563	3,700
Hazardous/low-level mixed waste	76	25	46	103	267
Low-level waste	57,000	20,000	31,000	35,260	140,000
Utilities and Energy Required for the Entire Period of Analysis (1998-2035)					
Water (millions of liters)	1,100	660	1,400	1186	8,000
Electricity (megawatt-hours)	46,000	27,000	81,000	116,000	600,000
Steam (millions of kilograms)	340	195	520	650	3,600
Diesel fuel (thousands of liters)	230	180	2,300	2760	22,000

- a. In the event that fuel receipts are less than those reported in Chapter 1, the values in this table that report impacts over the entire period of analysis would be less. Instructions for scaling impacts are provided in the appropriate Chapter 4 tables that provide input to this table.
- b. In the calculation of preferred alternative impacts, all the HEU/LEU oxides and silicides requiring resizing or special packaging have been accounted for in the melt and dilute technology even though a very small percentage would be conventionally processed. On the other hand, the loose-uranium-oxide-in-cans preferred alternative impacts do consider that 60 percent would be conventionally processed and the remaining 40 percent would be melted and diluted.
- c. MEI = maximally exposed offsite individual.
- d. Reflects current reactor-area emissions (including two SNF wet basins).
- e. LCF = latent cancer fatality.
- f. To calculate an annual impact, divide a number by 38. To calculate an impact for a given duration, multiply the annual impact by the duration in years. For example, the annual dose to the MEI from the preferred alternative would be 0.005 mrem (0.17/38). The estimated dose to the MEI until a storage facility would be operational (18 years from now) would be 0.040 mrem (0.005×8).

4.1.2.1 No-Action Alternative

Under the No-Action Alternative, SRS would continue to receive shipments of SNF from foreign research reactors, domestic research reactors, and other DOE sites. DOE would store the fuel in the L-Reactor Disassembly Basin or the Receiving Basin for Offsite Fuels, in addition to the currently stored SNF, under continued wet storage, and would ship the non-aluminum-clad fuel from these basins offsite. DOE would maintain the wet storage basins, performing upgrades as necessary to maintain proper water quality. The continued long-term underwater storage of aluminum-based SNF could lead to increased corrosion with increased environmental, health, and safety vulnerabilities. The No-Action Alternative consists of cases A8, B8, C8, D8, E8, and F8 (Table 4.1-27).

4.1.2.2 Minimum Impact Alternative

The identification of the Minimum Impact Alternative required both quantitative and quantitative analyses. The first step identified the minimum-impact technology for each fuel group for each analytical parameter (e.g., volume of high-level waste, air concentrations). However, the selection process often resulted in a combination of high and low impacts among parameters for a specific fuel group-technology combination cases; in other words, no clearly identified "best" or "worst" configuration was identified. Therefore, the second step was a qualitative examination of trends in configurations of cases that identified overall minimum impacts. Human health effects and environmental pollution impacts received slightly greater weight than consumption of natural resources or waste disposal space. In addition, impacts to the general public received slightly greater weight than those to SRS workers. The analysis indicates that cases A1, B1, C1, D3, E2, and F2 would provide minimum impacts (Table 4.1-28). Although other analysts could select different cases, DOE believes that the range

of impacts from reasonable choices of minimum-impact scenarios would be small and that the impacts of this combination would be representative of the lower bound of impacts from the proposed action.

4.1.2.3 Direct Disposal Alternative

This alternative combines the New Packaging and the Conventional Processing Technologies. Materials Test Reactor-like fuels and HEU/LEU Oxides and Silicides (except the failed and sectioned fuels) would be treated using the Direct Disposal/Direct Co-Disposal technology and placed in the Transfer and Storage Facility with a minimum of treatment (e.g., cold-vacuum drying and canning). The repackaging of the higher actinide targets and non-aluminum-clad fuels in the Transfer and Storage Facility would use the Repackage and Prepare to Ship technology. The uranium and thorium metal fuel, loose uranium oxide in cans, and failed and sectioned fuel from the HEU/LEU Oxides and Silicides fuel group would be treated using the Conventional Processing Alternative to alleviate the potential health and safety vulnerabilities discussed in Section 2.4.3.2 and because this material probably would not be suitable for placement in a geologic repository if treated with the Direct Disposal/Co-Disposal option. Therefore, the Direct Disposal alternative consists of cases A7, B1, C1, D7, E2, and F2 (Table 4.1-29).

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4.1.2.4 Preferred Alternative

DOE proposes to implement several of the technologies identified in Section 2.2 to manage spent nuclear fuel at SRS. These technologies are Melt and Dilute, Conventional Processing, and Repackage and Prepare to Ship. Each of these technologies would treat specific groups of spent nuclear fuel, as described below. The technology and fuel group combinations form DOE's Preferred Alternative in this EIS. The configuration of this preferred alternative is identified in Table 4.1-30.

Table 4.1-27. Fuel group and technology combination that compose the No-Action Alternative.

Fuel group	1	2	3	4	5	6	7	8
	Prepare for Direct Co-Disposal	Repackage and Prepare to Ship	Melt and Dilute	Mechanical Dilution	Vitrification Technologies	Electro-metallurgical Treatment	Conventional Processing	Continued Wet Storage
A. Uranium and Thorium Metal Fuels	-	-	-	-	-	-	-	Yes
B. Materials Test Reactor-like Fuels	-	-	-	-	-	-	-	Yes
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	-	-	-	-	-	-	-	Yes
D. Loose Uranium Oxide in Cans	-	-	-	-	-	-	-	Yes
E. Higher Actinide Targets	-	-	-	-	-	-	-	Yes
F. Non-Aluminum-Clad Fuels ^a	-	-	-	-	-	-	-	Yes

a. The environmental impacts of this case were analyzed in the Programmatic SNF EIS (DOE 1995b).
 HEU = highly enriched uranium.
 LEU = low enriched uranium.

Table 4.1-28. Fuel group and technology combination that compose the Minimum Impact Alternative.

Fuel group	1	2	3	4	5	6	7	8
	Prepare for Direct Co-Disposal	Repackage and Prepare to Ship	Melt and Dilute	Mechanical Dilution	Vitrification Technologies	Electro-metallurgical Treatment	Conventional Processing	Continued Wet Storage
A. Uranium and Thorium Metal Fuels	Yes	-	-	-	-	-	-	-
B. Materials Test Reactor-like Fuels	Yes	-	-	-	-	-	-	-
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	Yes	-	-	-	-	-	-	-
D. Loose Uranium Oxide in Cans	-	-	Yes	-	-	-	-	-
E. Higher Actinide Targets	-	Yes	-	-	-	-	-	-
F. Non-Aluminum-Clad Fuels ^a	-	Yes	-	-	-	-	-	-

a. The environmental impacts of this case were analyzed in the Programmatic SNF EIS (DOE 1995b).
 HEU = highly enriched uranium.
 LEU = low enriched uranium.

Table 4.1-29. Fuel group and technology combination that compose the Direct Disposal Alternative.

Fuel group	Prepare for Direct Disposal							
	1	2	3	4	5	6	7	8
	Co-Disposal	Repackage and Prepare to Ship	Melt and Dilute	Mechanical Dilution	Vitrification Technologies	Electro-metallurgical Treatment	Conventional Processing	Continued Wet Storage
A. Uranium and Thorium Metal Fuels	-	-	-	-	-	-	Yes	-
B. Materials Test Reactor-like Fuels	Yes	-	-	-	-	-	-	-
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	Yes	-	-	-	-	-	Yes ^a	-
D. Loose Uranium Oxide in Cans	-	-	Yes	-	-	-	Yes ^b	-
E. Higher Actinide Targets	-	Yes	-	-	-	-	-	-
F. Non-Aluminum-Clad Fuels ^a	-	Yes	-	-	-	-	-	-

- a. For failed or sectioned Oak Ridge Reactor fuel, High-Flux Isotope Reactor fuel, and Tower Shielding Reactor fuel, Heavy Water Components Reactor fuel, and Mark-42 targets.
- b. For Sterling Forest Oxide fuel.
- c. The environmental impacts of this case were analyzed in the Programmatic SNF EIS (DOE 1995b).
- HEU = highly enriched uranium.
LEU = low enriched uranium.

Table 4.1-30. Fuel group and technology combination that compose the Preferred Alternative.

	1							
	Prepare for Direct Co-Disposal		Repackage and Prepare to Ship		Melt and Dilute		Mechanical Dilution	
Fuel group								
	1	2	3	4	5	6	7	8
A. Uranium and Thorium Metal Fuels	-	-	-	-	-	-	Yes	-
B. Materials Test Reactor-like Fuels	-	-	Yes	-	-	-	-	-
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	-	-	Yes	-	-	-	Yes ^a	-
D. Loose Uranium Oxide in Cans	-	-	Yes	-	-	-	Yes ^b	-
E. Higher Actinide Targets	-	-	-	-	-	-	-	Yes ^c
F. Non-Aluminum-Clad Fuels ^c	-	Yes	-	-	-	-	-	-
TC								
TC	a. For failed or sectioned Oak Ridge Reactor fuel, High-Flux Isotope Reactor fuel, and Tower Shielding Reactor fuel, Heavy Water Components Test Reactor fuel, and Mark-42 targets.							
	b. For Sterling Forest Oxide fuel.							
	c. The environmental impacts of this case were analyzed in the Programmatic SNF EIS (DOE 1995b). HEU = highly enriched uranium. LEU = low enriched uranium. NA = not applicable; not decided in this EIS.							

4.1.2.4.1 Melt And Dilute

DOE has identified the Melt and Dilute process as the preferred method of treating most (about 97 percent by volume or about 32,000 MTRE) of the aluminum-based SNF considered in this EIS. DOE will continue to pursue a research and development program leading to a demonstration of the technology in FY 2001 using full-size irradiated research reactor spent nuclear fuel assemblies. With a successful demonstration of the technology, DOE expects to have ready a treatment facility to perform production melt and dilute operations in FY 2008. DOE will ensure the continued availability of SRS conventional processing facilities until we have successfully demonstrated implementation of the Melt and Dilute treatment technology.

The fuel proposed for the preferred Melt and Dilute technology includes the Material Test Reactor-like fuel, most of the Loose Uranium Oxide in Cans fuel, and most of the HEU/LEU Oxide and Silicide fuel. Exceptions are the uranium and thorium fuel, failed and sectioned oxide and silicide fuel, some loose uranium oxide in cans fuel, the Higher Actinide Targets, and non-aluminum-clad fuel.

If DOE identifies any health or safety concerns involving any aluminum-based SNF prior to the melt and dilute facility becoming operational, DOE could use F and H Canyons to stabilize the material of concern, if the canyons were not de-commissioned.

4.1.2.4.2 Conventional Processing

DOE has identified conventional processing to manage a relatively small volume of aluminum-based SNF at the SRS (about 3 percent by volume; less than 3,000 MTRE) that presents a potential health and safety vulnerability or is in a form that may be unacceptable for placement in a geologic repository. That SNF includes the Experimental Breeder Reactor-II fuel, the Sodium Reactor Experiment fuel, the Mark-42 targets and the core filter block from the Uranium and Thorium Metal fuel group; the failed or sectioned Tower Shielding Reactor, High Flux Isotope Reactor, Oak Ridge Reactor, and Heavy

Water Components Test Reactor fuels and a Mark-14 target from the HEU/LEU Oxides and Silicides fuel group; and the Sterling Forest Oxide (and any other powdered/oxide fuel that may be received at SRS while H Canyon is still in operation) from the Loose Uranium Oxide in Cans fuel group.

4.1.2.4.3 Repackaging

DOE proposes to repackage the non-aluminum-clad fuel at SRS and transfer the material to dry storage. DOE would transfer the non-aluminum-clad fuel to that facility for storage pending off-site shipment. DOE expects transfer operations would begin in time to support closing the Receiving Basin for Offsite Fuels by 2007. Depending on receipt schedules for research reactor fuels and the operating schedule for the melt and dilute facility, DOE could deinventory the Receiving Basin for Offsite Fuels and move any remain fuel to the Building 105-L wet basin prior to packaging the fuel for dry storage.

The Preferred Alternative would include cases A7, B3, C3, D3, E2, and F2 (Table 4.1-30).

4.1.2.4.4 Continued Wet Storage

DOE proposed to maintain the higher actinide target fuel group in continued wet storage pending decisions on final disposition.

4.1.2.5 Maximum Impact Alternative

This alternative provides the upper bound on the range of impacts from potential configurations. It would provide conventional processing for all SNF except the higher actinide targets and the non-aluminum-clad fuels selected for offsite shipment and deemed inappropriate for conventional processing. The higher actinide targets would be repackaged for potential offsite shipment and dry-stored until DOE made a decision regarding their disposition. The non-aluminum-clad fuels would be packaged for shipment and dry stored until they were ready for shipment to the Idaho National Engineering and Environmental Laboratory.

Table 4.1-31. Fuel group and technology combination that compose the Maximum Impact Alternative.

	1 2 3 4 5 6 7 8							
	Prepare for Direct Co-Disposal	Repackage and Prepare to Ship	Melt and Dilute	Mechanical Dilution	Vitrification Technologies	Electro- metallurgical Treatment	Conventional Processing	Continued Wet Storage
TC Fuel group								
A. Uranium and Thorium Metal Fuels	-	-	-	-	-	-	Yes	-
B. Materials Test Reactor-like Fuels	-	-	-	-	-	-	Yes	-
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	-	-	-	-	-	-	Yes	-
D. Loose Uranium Oxide in Cans	-	-	-	-	-	-	Yes	-
E. Higher Actinide Targets	-	Yes	-	-	-	-	Yes ^a	-
F. Non-Aluminum-Clad Fuels ^b	-	Yes	-	-	-	-	-	-
TC								
a. The environmental impacts of processing Mark-18 targets was analyzed in the Interim Management of Nuclear Materials Final Environmental Impact Statement (DOE 1995a).								
b. The environmental impacts of this case were analyzed in the Programmatic SNF EIS (DOE 1995b). HEU = highly enriched uranium. LEU = low enriched uranium.								

Analyses of the maximum impact alternative are conservative in that they assume that the entire SNF inventory would be processed in the canyons, which would produce the greatest impacts of all the treatment options. No credit is taken for discontinuing use of the canyons and processing some of the inventory in a new treatment facility. The Conventional Processing Alternative would include cases A7, B7, C7, D7, E2, and F2 (Table 4.1-31). DOE believes that this combination would provide an upper bound on impacts.

4.2 Accident Analysis

This section summarizes risks to the public and workers from potential accidents associated with the technology options for SNF management at the SRS.

An accident is a sequence of one or more unplanned events with potential outcomes that endanger the health and safety of workers and the public. An accident can involve a combined release of energy and hazardous materials (radiological or chemical) that might cause prompt or latent health effects. The sequence usually begins with an initiating event, such as a human error followed by an explosion, or an earthquake followed by structural failure. A succession of other events, such as a ventilation system failure, that are dependent or independent of the initial event, could affect the magnitude of the accident and the materials released. Initiating events fall into three categories:

- *Internal initiators* normally originate in and around the facility but are always a result of facility operations (equipment or structural failures, human errors, internal flooding).
- *External initiators* are independent of facility operations and normally originate outside the facility (aircraft crashes, nearby explosions, and toxic chemical releases at nearby facilities that affect worker performance); some can affect the ability of the facility to maintain confinement of hazardous materials because of structural damage.

- *Natural phenomena initiators* are natural occurrences that are independent of facility operations and of *events* at nearby facilities or operations (earthquakes, high winds, floods, lightning, snow). Natural phenomena initiators could affect external facilities, which could in turn affect other facilities and compound the progression of the accident.

Table 4.2-1 summarizes the estimated impacts to workers and the public from potential accidents for each SNF technology option. All the options would require the use of the Receiving Basin for Offsite Fuels and the L-Reactor Disassembly Basin. All except Continued Wet Storage would require the construction and operation of a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility.

The table lists the impacts of potential accidents in relation to the phases required to implement each option. They list only the accident with the worst impacts based on the maximally exposed offsite individual. Appendix D contains details of the impacts of other postulated accidents. Table 4.2-1 lists potential accident consequences as latent cancer fatalities, without consideration of the accident's probability. The calculation of latent cancer fatalities from population dose is performed in the same manner as for non-accident radiological health effects presented in section 4.1.1.3.1.

DOE estimated impacts to three receptors: (1) an uninvolved worker 2,100 feet (640 meters) from the accident location as discussed in DOE (1994), (2) the maximally exposed individual at the SRS boundary, and (3) the offsite population in an area within 50 miles (80 kilometers).

Many of the analysis results presented in Table 4.2-1 are substantially different from those given in the draft EIS. DOE has continued to conduct research and development, including accident analyses, to determine the feasibility of implementing technologies and the potential health and safety consequences of doing so. In some cases design changes have been

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Table 4.2-1. Estimated maximum consequence accident for each technology.

Option	Accident Frequency	Consequences			
		Noninvolved Worker (rem)	MEI (rem)	Offsite Population (person-rem)	Latent Cancer Fatalities
Continued Wet Storage (No Action) ^a					
RBOF (high wind-induced criticality)	Once in 26,000 years	13	0.22	12,000	6.2
L-Reactor basin (basin-water draindown)	Once in 500 years	0.014	0.016	(b)	(b)
Direct Co-Disposal					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Repackage and Prepare to Ship					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Conventional Processing					
Processing phase in F/H Canyons (coil and tube failure)	Once in 14,000 years	13	1.3	78,000	39
Melt and Dilute					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Melt and dilute phase (earthquake induced spill with loss of ventilation)	Once in 200,000 years	30	0.5	21,000	10
Mechanical Dilution					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Mechanical dilution phase (criticality with loss of ventilation)	Once in 33,000 years	0.71	0.074	3,000	1.5
Vitrification Technologies					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Vitrification phase (earthquake-induced release with loss of ventilation)	Once in 200,000 years	0.10	0.0017	71	0.035
Electrometallurgical Treatment					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Electrometallurgical phase (metal melter earthquake induced spill with loss of ventilation)	Once in 200,000 years	30	0.5	21,000	10

MEI = Maximally Exposed Individual.

RBOF = Receiving Basin for Offsite Fuels.

a. All alternatives would use RBOF and the L-Reactor Disassembly Basin; therefore, accidents in these facilities are possible for each technology.

b. Not available.

TC

considered specifically to reduce the potential for accidents with adverse consequences. During that process, assumptions about the design and operation of the proposed technologies have changed. Changes in the assumptions have resulted in changes in the outcome of the accident analyses. Details concerning the analyses are found in Appendix D of this EIS.

For all of the accidents, there is a potential for injury or death to involved workers in the vicinity of the accident. In some cases, the impacts to the involved worker would be greater than to the noninvolved worker. However, prediction of latent potential health effects becomes increasingly difficult to quantify as the distance between the accident location and the receptor decreases because the individual worker exposure cannot be precisely defined with respect to the presence of shielding and other protective features. The worker also may be acutely injured or killed by physical effects of the accident itself. DOE identified potential accidents through a detailed hazard assessment and estimated impacts using the AXAIRQ computer model (Simpkins 1995a,b), as discussed in Appendix D.

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Results of accident calculations listed in Table 4.2-1 have been updated since the Draft EIS to incorporate evolution of the technology alternatives and to incorporate information that was not available at the time the Draft EIS was prepared.

4.3 Construction Impacts

This section describes environmental impacts that could result from construction activities associated with SNF management at SRS. These activities would include the construction of a Transfer and Storage Facility under the New Packaging Technology or the construction of a Transfer, Storage, and Treatment Facility under the New Processing Technology or Conventional Processing. DOE does not expect such construction activities to have appreciable impacts on geologic resources, groundwater, traffic, transportation, or cultural resources, as explained below

4.3.1 GEOLOGY AND GROUNDWATER

DOE would confine the construction of new facilities to previously disturbed and developed areas and, therefore, expects little or no environmental impacts to the geologic resources of the area. Neither the construction nor the operation of the proposed Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would affect groundwater in the area. The proposed DOE action to remove stored fuels from existing basins would eliminate a potential source of environmental releases (leaks from wet basins). The Transfer and Storage Facility or Transfer, Storage, and Treatment Facility could include the capability to perform wet receipt and unloading of SNF.

4.3.2 TRAFFIC AND TRANSPORTATION

DOE would transport construction materials, wastes, and excavated materials associated with building the proposed facilities both on and off SRS. These activities would result in increases in the operation of personal vehicles by construction workers, commercial truck traffic, and traffic associated with the daily operations of SRS. However, increases in worker and materials traffic would be small in comparison to existing traffic loads. Increased traffic congestion would be minimal.

4.3.3 CULTURAL RESOURCES

As discussed in Section 3.6, activities associated with the proposed action and alternatives for SNF management at SRS that could affect cultural resources would be the use of the three candidate sites for the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility. These sites are in reactor areas (L, C, and P) within 100 to 400 yards (91 to 366 meters) of the reactor buildings. The Savannah River Archaeological Research Program has not examined these sites. The Site Use Program, which requires a permit for clearing land on the SRS, usually initiates archaeological investigations. DOE would direct an investigation of the selected site before starting facility design and construction. Although there were homesites at

or near the proposed facility sites in C and L Areas, the likelihood of historic resources surviving the construction of the reactors in the early 1950s, before the enactment of regulations to protect such resources would be small (Sassaman 1997).

The potential for the presence of prehistoric sites in the candidate locations also is limited. The L-Area site is in archaeological site density Zone 3, which has the least potential for prehistoric sites of significance. The C-Area site is in Zones 2 and 3 and has more potential. Zone 2 includes areas of moderate archaeological site density. The P-Area site is in Zone 2. However, as with any historic sites, reactor construction activities probably destroyed or severely damaged prehistoric deposits. DOE would direct an examination of the selected location for prehistoric resources before starting the design and construction of the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility (Sassaman 1997).

4.3.4 SURFACE WATER RESOURCES

Construction at SRS must comply with the requirements of South Carolina stormwater management and sediment reduction regulations, which became effective in 1992 as part of the Clean Water Act. These regulations and their associated permits require DOE to prepare erosion and sediment control plans for all projects, regardless of the land area. Runoff from the construction site would be part of a stormwater management and sedimentation control plan to minimize potential discharges of silts, solids, and other contaminants to surface-water streams. Effective January 2, 1997, the South Carolina Department of Health and Environmental Control (SCDHEC) approved General Permit coverage for stormwater management and sediment reduction at the SRS (SCDHEC 1996). Although the General Permit does not exempt any land-disturbing and construction activities from the requirements of State stormwater management and sediment control regulations, it does preclude the necessity of SCDHEC plan review and approval for land disturbing and construction activities at the SRS.

Before beginning construction, DOE would develop erosion and sediment control plans for the planned facilities. After construction and depending on the location of the construction site, the *SRS Stormwater Pollution Prevention Plan* (WSRC 1993), which is a requirement of the general NPDES stormwater permit covering industrial activities (Permit SCR000000), would include applicable erosion and sediment control measures; inclusion in the plan would not be necessary if the facility to be constructed was in the drainage area of a stormwater collection system permitted as part of NPDES Permit SC0000175.

4.3.5 AIR RESOURCES

The potential construction of facilities for the management of SNF would cause emissions of fugitive dust (particulate matter) from land-clearing activities and exhaust emissions from construction equipment (earth-moving vehicles, diesel generators). DOE has considered such impacts for activities at SRS that were similar in facility size and application and concluded that impacts to air quality would be minimal (DOE 1995a,b) and would have no effect on SRS compliance with state and Federal ambient air quality standards. Concentrations of pollutants emitted during construction activities would be at least an order of magnitude less than the South Carolina ambient air quality standards.

4.3.6 ECOLOGICAL RESOURCES

DOE is considering three brown field sites for the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility, if they are not constructed in a renovated reactor: C Area, L Area, and P Area. As noted in Section 3.4, the sites would encompass approximately 60,700 square meters (15 acres), including the main building and land required for ancillary facilities. The Treatment Facility could also be constructed on a previously disturbed site inside the F-Area or H-Area fences.

All construction activity for the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would take place within the

boundary of one of the three reactor areas in an already-developed brownfield area. Undeveloped portions of the three proposed sites provide some low-quality wildlife habitat.

Construction of the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would involve the movement of workers and construction equipment, and would be associated with relatively loud noises from earth-moving equipment, portable generators, pile-driving equipment, pneumatic tools, drills, hammers, and the like. Although noise levels in construction areas could be as high as 110 dBA, these high local noise levels would not extend far beyond the boundaries of the project site.

Table 4.3-1 gives the attenuation of construction noise over relatively short distances. At 120 meters (400 feet) from the construction site, construction noises would range from approximately 60 to 80 dBA. Golden et al. (1980) suggest that noise levels higher than 80 to 85 dBA are sufficient to startle or frighten birds and small mammals. Thus, there would be minimal

Potential for disturbing birds and small mammals outside a 120-meter radius from the construction site.

Although noise levels would be relatively low outside the immediate area of construction, the combination of construction noise and human activity probably would displace small numbers of animals (e.g., songbirds and small mammals) that could forage, feed, nest, rest, or den in the area. Construction-related disturbances are likely to create impacts to wildlife that would be small, temporary (approximately 24 months), and localized. Some animals could be driven from the area permanently, while others could become accustomed to the increased noise and activity and return to the area. Species likely to be affected (e.g., gray squirrel, opossum, white-tailed deer) are common to ubiquitous in these areas. Construction would not disturb any threatened or endangered species, would not degrade any critical or sensitive habitat, and would not affect any jurisdictional wetlands.

Table 4.3-1. Peak and attenuated noise (in dBA) levels expected from operation of construction equipment.^a

Source	Noise level (peak)	Distance from source			
		50 feet ^b	100 feet	200 feet	400 feet
Heavy trucks	95	84-89	78-83	72-77	66-71
Dump trucks	108	88	82	76	70
Concrete mixer	105	85	79	73	67
Jackhammer	108	88	82	76	70
Scraper	93	80-89	74-82	68-77	60-71
Dozer	107	87-102	81-96	75-90	69-84
Generator	96	76	70	64	58
Crane	104	75-88	69-82	63-76	55-70
Loader	104	73-86	67-80	61-74	55-68
Grader	108	88-91	82-85	76-79	70-73
Dragline	105	85	79	73	67
Pile driver	105	95	89	83	77
Fork lift	100	95	89	83	77

a. Source: Golden et al. (1980).

b. To convert feet to meters, multiply by 0.3048.

4.3.7 IMPACTS FROM RENOVATING AN EXISTING FACILITY

4.3.7.1 Waste Generation

As discussed in Section 2.3.2.3, DOE could locate the Transfer, Storage, and Treatment Facility in a renovated reactor area, such as the 105-L facility. This would require decontamination and removal of components and systems and subsequent construction activities inside the reactor building and would result in impacts that would not occur during the construction of a virgin facility. Impacts would include generation of radioactive waste during decontamination, removal and construction. DOE has estimated that decontamination and removal and construction activities would result in the generation of approximately 476 m³ of low-level waste over the total duration of the activities (WSRC 1998). Eventual decontamination and decommissioning (D&D) of the Transfer, Storage, and Treatment Facility (either stand-alone or in a renovated reactor facility) also would result in generation of radioactive waste.

4.3.7.2 Worker Health

DOE could locate the Transfer, Storage, and Treatment Facility in a renovated reactor area, such as the 105-L facility. This would require decontamination and removal of components and systems and subsequent construction activities inside the reactor building and would result in impacts that would not occur during the construction of a virgin facility. Impacts would include radiation exposure of workers performing these activities. The decontamination and removal and construction activities would result in a total collective worker radiation dose of 32 person-rem, based on 54 total workers and a duration of 1 year to complete all activities (Nathen 1998). The collective worker dose is

estimated to result in 1.3×10^{-3} latent cancer fatalities. Eventual decontamination and decommissioning (D&D) of the Transfer, Storage, and Treatment Facility (either stand-alone or in a renovated reactor facility) also would result in radiation exposure of D&D workers.

4.3.8 SOCIOECONOMIC IMPACTS

The implementation of the alternatives discussed in this EIS could result in the construction and operation of a Transfer and Storage Facility or a Transfer, Storage and Treatment Facility, which could in turn cause incremental socioeconomic impacts in the SRS area. Section 2.3.2 discusses the construction and operation of the Transfer and Storage Facility. Its construction would cost an estimated \$200 million. A 2-year construction period would result in a short-term increase of fewer than 500 jobs in the region, approximately 75 percent of which would be in construction. This would be an increase in construction jobs of approximately 2 percent (from about 16,000) and an increase of considerably less than 1 percent in total employment for the region (REMI 1995). After the 2-year period, employment would return back to its previous equilibrium. The small temporary increases in employment would not present significant impacts to the regional economy, services, or infrastructure.

DOE would construct the treatment phase of the Transfer, Storage, and Treatment Facility after the Transfer and Storage phase was constructed; the construction periods would not overlap. The treatment phase would require less effort to construct and would employ fewer construction employees.

None of these construction activities would significantly increase regional employment or population, and socioeconomic impacts would be negligible.

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References

- Arnett, M. W., 1996, *Savannah River Site Environmental Data for 1995*, WSRC-TR-96-0077, Westinghouse Savannah River Company, Aiken, South Carolina.
- Arnett, M. W., and A. R. Mamatey, 1996, *Savannah River Site Environmental Report for 1995*, WSRC-TR-96-0075, Westinghouse Savannah River Company, Aiken, South Carolina.
- Bickford, W. E., J. F. Krupa, D. L. McWhorter, M. E. Dupont, 1997, *Savannah River Site Spent Nuclear Fuel Environmental Impact Statement Engineering Data Book for Routine Releases*, WSRC-TR-97-0044, Westinghouse Savannah River Company, Aiken, South Carolina.
- CDC (Centers for Disease Control and Prevention), 1996, *Monthly Vital Statistics Report*, Volume 45, No. 3 Supplement, "Advance Report of Final Mortality Statistics, 1994," U.S. Public Health Service, Department of Health and Human Services, Washington, D.C., September 30.
- Choi, A. S., 1992, "Material Balance Tables for DWPF Radioactive Runs with Batch 1 Sludge/Supernate Feed," Revision 1, Draft, interoffice memorandum to L.F. Landon, Westinghouse Savannah River Company, Aiken, South Carolina, November 6.
- DOE (U.S. Department of Energy), 1994, *DOE Standard - Guidance for Preparation of DOE 5480.22 (TSR) and DOE 5480.23 (SAR) Implementation Plans*, DOE-STD-3011-1994, Washington, D.C.
- DOE (U.S. Department of Energy), 1995a, *Final Environmental Impact Statement, Interim Management of Nuclear Materials*, DOE/EIS-0220, Savannah River Site, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1995b, *Final Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement*, DOE/EIS-0203, Idaho Operations Office, Idaho Falls, Idaho.
- DOE (U.S. Department of Energy), 1996, *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel*, DOE/EIS-0218F, Washington, D.C.
- DOE (U.S. Department of Energy), 1997, *Final Environmental Impact Statement, Shutdown of the River Water System at the Savannah River Site*, DOE/EIS-0268, Savannah River Operations Office, Aiken, South Carolina.
- TC | DOE (U.S. Department of Energy), 1999, *Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County Nevada*, DOE/EIS-0250D, Office of Civilian Radioactive Waste Management, Washington, D.C.
- Golden, J., R. P. Ouellette, S. Saari, P. N. Cheremisinoff, 1980, *Environmental Impact Data Book*, Ann Arbor Science Publishers, Inc., Ann Arbor, Michigan.
- Hamby, D. M., 1991, "LADTAP XL: An Improved Electronic Spreadsheet Version of LADTAP II." WSRC-RP-91-975, Westinghouse Savannah River Company, Aiken, South Carolina.

- Hamby, D. M., 1994, "Verification of the MAXIGASP and POPGASP Compute-Codes for Environmental Dose Assessment," WSRC-RP-94-522, Westinghouse Savannah River Company, Aiken, South Carolina.
- HNUS (Halliburton NUS Corporation), 1994a, *Transportation Radiological Analysis - Programmatic Spent Nuclear Fuel Management Environmental Impact Statement*, Aiken, South Carolina.
- HNUS (Halliburton NUS Corporation), 1994b, *Transportation Radiological Analysis Calculations for Interim Management of Nuclear Materials Environmental Impact Statement*, Aiken, South Carolina.
- McWhorter, D. L., 1997, "SRS SNF Management EIS, SNF Processing Durations (U)," interoffice memorandum to C. B. Shedrow, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina, May 28.
- Mayer, J. J., and L. D. Wike, 1997, *SRS Urban Wildlife Environmental Information Document*, WSRC-TR-97-0093, Westinghouse Savannah River Company, Aiken, South Carolina.
- Nathen, S., 1998, "Construction Dose Estimate – Operational Data Needs," interoffice memorandum to K. Waltzer (U.S. Department of Energy), Westinghouse Savannah River Company, Aiken, South Carolina, May 4.
- NCRP (National Council on Radiation Protection and Measurements), 1993, *Limitation of Exposure to Ionizing Radiation*, Report No. 116, Washington, D.C.
- Neuhauser, K. S., and F. L. Kanipe, 1992, *RADTRAN 4: Volume 3 User Guide*, SAND89-2370; TTC-0943; UC-772, Sandia National Laboratories, Albuquerque, New Mexico.
- NRC (U.S. Nuclear Regulatory Commission, 1975, *Regulation Guide 1.13 - Spent Fuel Storage Facility Design Basis*, Office of Standards Development, Washington, D.C.
- NRC (U.S. Nuclear Regulatory Commission, 1977, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 20, Appendix I*, Regulatory Guide 1.109, Revision 1, Washington, D.C.
- REMI (Regional Economic Models, Inc.), 1995, *Economic and Demographic Forecasting and Simulation Model*, Amherst, Massachusetts.
- Sassaman, K., 1997, "Archaeological Review of Proposed Spent Fuel Facility at P-Reactor," memorandum to L. S. Moore (Halliburton NUS Corporation), Savannah River Archaeological Research Program, South Carolina Institute of Archaeology and Anthropology, Aiken, South Carolina.
- SCDHEC (South Carolina Department of Health and Environmental Control), 1996, "General Permit for Land Disturbing/Construction Activities SRS/Aiken County," letter to Westinghouse Savannah River Company, December 11.
- Simpkins, A. A., 1995a, *Verification of AXAIRQ*, WSRC-RP-95-708, Westinghouse Savannah River Company, Savannah River Technology Center, Aiken, South Carolina.
- Simpkins, A. A., 1995b, *Verification of AXAIRQ*, WSRC-RP-95-709, Westinghouse Savannah River Company, Savannah River Technology Center, Aiken, South Carolina.

- Simpkins, A. A., 1996, "Spent Nuclear Fuel EIS Routine Release Environmental Dosimetry Calculations," SRT-ETS-960149, memorandum to C. B. Shedrow, Westinghouse Savannah River Company, Aiken, South Carolina.
- WSRC (Westinghouse Savannah River Company), 1993, *SRS Stormwater Pollution Prevention Plan*, WSRC-IM-93-28, Savannah River Site, Aiken, South Carolina.
- WSRC (Westinghouse Savannah River Company), 1995, *Receiving Basin for Off-Site Fuel and Resin Regeneration Facility Safety Analysis Report*, draft, WSRC-SA-11 (WSRC-TR-95-0054), Savannah River Site, Aiken, South Carolina.
- WSRC (Westinghouse Savannah River Company), 1996a, *Basis for Interim Operation (BIO) for the L-Reactor Facility (U)*, Site Configuration and Safety Services Department, WSRC-TR-95-0054, Aiken, South Carolina.
- WSRC (Westinghouse Savannah River Company), 1996b, *Safety Analysis - 200 Area Savannah River Site H-Canyon Operations*, H-Canyon SAR Addendum, DPSTSA-200-10 Supp-5, Addendum 6, Revision 0, Aiken, South Carolina.
- WSRC (Westinghouse Savannah River Company), 1998, *Estimate Summary Level Report - 105-L Alternative for Spent Nuclear Fuel Transfer and Storage Services*, Log Number 98-01-18, Aiken, South Carolina.

CHAPTER 5. CUMULATIVE IMPACTS

The Council on Environmental Quality (CEQ) regulations implementing the procedural provisions of the National Environmental Policy Act (NEPA) define cumulative impacts as the impacts on the environment which result from the incremental impact of the action when added to other past, present, and reasonably foreseeable future actions regardless of what agency (Federal or non-Federal) or person undertakes such other actions (40 CFR 1508.7). The cumulative impacts analysis presented in this section is based on the incremental actions associated with the maximum impact alternative for spent nuclear fuel (SNF) management at the Savannah River Site (SRS), other actions associated with onsite activities, and offsite activities with the potential for related environmental impacts. Although it is unlikely that the maximum impact alternative would be implemented to manage SNF at SRS, it was used to estimate cumulative impacts to ensure a conservative analysis. In accordance with a handbook recently prepared by CEQ (1997), the U.S. Department of Energy (DOE) identified the resource areas in which SNF management could add to the impacts of past, present, and reasonably foreseeable actions within the project impact zones as defined by CEQ (1997).

Based on an examination of the environmental impacts of direct and indirect SNF management actions coupled with DOE and other agency actions, it was determined that cumulative impacts for the following areas need to be presented: (1) air resources; (2) water resources; (3) public and worker health; (4) waste generation; (5) utilities and energy consumption; and (6) socioeconomic. Discussion of cumulative impacts for the following resources is omitted because impacts from the proposed SNF management activities would be so small that their potential contribution to cumulative impacts would be negligible: geologic resources, ecological resources, aesthetic and scenic resources, cultural resources, and traffic.

For determining the impact to air, water, human health, waste generation, utilities and energy, and socioeconomic resources from commercial and Federal nuclear facilities, the 50-mile (80-kilometer) radius surrounding SRS was selected as the project impact zone. For aqueous releases, the downstream population that uses the Savannah River as its source of drinking water was included in the project impact zone.

Nuclear facilities within a 50-mile radius of SRS include Georgia Power's Plant Vogtle Electric Generating Plant across the river from SRS; Chem-Nuclear Inc., a commercial low-level waste burial site just east of SRS; and Starmet CMI, Inc. (formerly Carolina Metals), located southeast of SRS, which processes uranium-contaminated metals. Radiological impacts from the operation of the Vogtle Electric Generating Plant, a two-unit commercial nuclear power plant are minimal, but DOE has factored them into the analysis. The South Carolina Department of Health and Environmental Control Annual Report (SCDHEC 1995) indicates that operation of the Chem-Nuclear Services facility and the Starmet CMI facility do not noticeably impact radiation levels in air or liquid pathways in the vicinity of SRS. Therefore, they are not included in this assessment.

The counties surrounding SRS have numerous existing (e.g., textile mills, paper product mills, and manufacturing facilities) and planned (e.g., Bridgestone Tire) industrial facilities with permitted air emissions and discharges to surface waters. Because of the distances between SRS and the private industrial facilities, there is little opportunity for interactions of plant emissions, and no major cumulative impact on air or water quality. Construction and operation of Bridgestone Tire and Hankook Polyester facilities could affect the regional socioeconomic cumulative impacts.

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Additional offsite facilities with the potential to affect the nonradiological environment include South Carolina Electric and Gas Company's Urquhart Station. Urquhart Station is a three-unit, 250-megawatt, coal- and natural-gas-fired steam electric plant in Beech Island, South Carolina, located about 32 river kilometers (20 river miles) north of SRS. Because of the distance between SRS and the Urquhart Station and the regional wind direction frequencies, there is little opportunity for any interaction of plant emissions, and no significant cumulative impact on air quality.

impacts of consolidating the tritium activities currently the new Building 233-H and Building 234-H. Tritium extraction functions would be transferred to Tritium Extraction Facility. The overall impact would be to reduce the tritium facility complex net tritium emissions by up to 50 percent. Another positive effect of this planned action would be to reduce the amount of low-level radioactive job-control waste. Effects on other resources would be negligible. Therefore, impacts from the environmental assessment have not been included in this cumulative impacts analysis.

EC | DOE also evaluated the impacts from its own proposed future actions by examining impacts to resources and the human environment as shown in NEPA documentation related to SRS (see Section 1.6). Additional NEPA documents related to SRS that are considered in the cumulative impacts section include the following:

EC | ***Final Environmental Impact Statement - Interim Management of Nuclear Materials (DOE/EIS-0220)*** (DOE 1995a). DOE has begun implementation of the preferred alternatives for the nuclear materials discussed in the Interim Management of Nuclear Materials EIS. SRS baseline data in this chapter reflect projected impacts from implementation.

TC |

EC | ***Final Environmental Impact Statement for the Accelerator Production of Tritium at Savannah River Site (DOE/EIS-0270)*** (DOE 1999a). DOE has proposed an accelerator design (using helium-3 target blanket material) and an alternate accelerator design (using lithium-6 target blanket material). If an accelerator is built, it would be located at SRS. However, since the Record of Decision states the preferred alternative as use of an existing commercial light-water reactor, data from this EIS are not used.

TC |

EC | ***Environmental Assessment for the Tritium Facility Modernization and Consolidation Project at the Savannah River Site (DOE/EA-1222)*** (DOE 1997). This environmental assessment (EA) addresses the

Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement (DOE/EIS-0240) (DOE 1996). This cumulative impacts analysis incorporates the alternative of blending at SRS highly enriched uranium to 4 percent low-enriched uranium as uranyl nitrate hexahydrate as stated in the Record of Decision (61 FR 40619, August 5, 1996).

EC |

TC |

Final Environmental Impact Statement on Management of Certain Plutonium Residues and Scrub Alloy Stored at the Rocky Flats Environmental Technology Site (DOE/EIS-0277F) (DOE 1998). DOE proposes to process certain plutonium-bearing materials being stored at the Rocky Flats Environmental Technology Site. These materials are plutonium residues and scrub alloy remaining from nuclear weapons manufacturing operations formerly conducted by DOE at Rocky Flats. DOE has decided to remove the plutonium from certain residues that would be shipped from the Rocky Flats Environmental Technology Site to SRS for stabilization. The separated plutonium would be stored at SRS pending disposition decisions. Environmental impacts from using F Canyon to chemically separate the plutonium from the remaining materials at SRS are included in this section.

EC |

EC | ***Final Environmental Impact Statement for the Construction and Operation of a Tritium Extraction Facility at the Savannah River Site (DOE/EIS-0271)*** (DOE 1999b). As stated in the Record of Decision (64 FR 26369; 5/14/99), DOE will construct and operate a Tritium Extraction Facility on SRS to provide the capability to extract tritium from commercial light water reactor targets and targets of similar design. The purpose of the proposed action and alternatives evaluated in the EIS is to provide tritium extraction capability to support either accelerator or reactor production. Environmental impacts from the maximum processing option in this EIS are included in this section.

EC | ***Surplus Plutonium Disposition Final Environmental Impact Statement (DOE/EIS-0283)*** (DOE 1999c). This EIS analyzes the activities necessary to implement DOE's disposition strategy for surplus plutonium. In January 2000 DOE issued a Record of Decision selecting SRS as the site for all three disposition facilities: mixed-oxide fuel fabrication, plutonium immobilization, and plutonium pit disassembly and conversion. Impacts from these facilities are included in this section.

TC |

EC | ***Defense Waste Processing Facility Supplemental Environmental Impact Statement (DOE/EIS-0082-S)*** (DOE 1994). The selected alternative in the Record of Decision (ROD) was the completion and operation of the Defense Waste Processing Facility to immobilize high-level radioactive waste at the SRS. The facility is currently processing sludge from SRS high-level waste tanks. However, SRS baseline data is not representative of full DWPF operational impacts, including processing of salt and supernate from these tanks. Therefore, the DWPF data is listed separately.

L2-16
EC | ***Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel (DOE/EIS-0306D)*** (DOE 1999d). DOE has

published a draft environmental impact statement (64 FR 8553, 2/22/99) for treatment of sodium-bonded spent nuclear fuel. Two of the alternatives being evaluated in the Treatment and Management EIS are to process INEEL's sodium-bonded fuel inventory at SRS using the Plutonium-Uranium Extraction (PUREX) process and to use the Melt and Dilute facility being proposed in the EIS. Because processing at SRS is a reasonable alternative to processing at INEEL, it is being included in the Spent Nuclear Fuel Management EIS cumulative impact analysis. These methods could be used for the sodium-bonded spent nuclear fuel blanket assemblies currently in storage at INEEL. There are approximately 22.4 MTHM of Experimental Breeder Reactor-II (EBR-II) fuel blankets and 34.2 MTHM of Fermi-1 fuel blankets to be processed. This fuel would be declad before shipment to SRS. Because the decladding activities would occur at INEEL, the impacts of these decladding activities are not included in this chapter.

This EIS includes cumulative impacts of sodium-bonded spent nuclear fuel processing at the SRS based on data from the Draft Electrometallurgical Treatment EIS. Data used in this EIS are based on Purex processing at SRS, which is more conservative.

DOE is currently evaluating nuclear material disposition needs. Other material discussed for processing at SRS under the PNA include single-pass reactor SNF at Hanford, a small amount of damaged SNF at Idaho National Engineering and Environmental Laboratory (INEEL), classified fissile material metal parts at the Rocky Flats Environmental Technology Site (RFETS), and plutonium scrap at Hanford. Currently, DOE has no plan or proposal to transfer the single-pass reactor SNF at Hanford or the damaged SNF at INEEL to SRS so that material was not considered for the cumulative impacts under this EIS. In an amended Record of Decision for the *Final Environmental Impact Statement on Storage and Disposition of Surplus Fissile Material*,

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TC

	DOE decided to transfer classified metal from RFETS to SRS for stabilization and storage. DOE is considering transferring the plutonium scrap from Hanford to SRS for stabilization and storage pending appropriate National Environmental Policy Act review. As a result, DOE has included processing that material as part of the cumulative impacts for this EIS.	SNF from SRS for disposal would be completed by 2035.	
L2-16	DOE is continuing to evaluate the inventory of nuclear material at facilities throughout the DOE complex. DOE's Nuclear Material Integration initiative is one such recent effort that has identified material which could be processed at SRS. Although there are no current plans to process these materials at SRS, DOE considers it appropriate to include a qualitative estimate of impacts as part of the cumulative impacts for this EIS because it is not unforeseen that processing at SRS could occur.	The period of interest for the cumulative impacts analysis for this SNF EIS includes the potential construction and operation of the Tritium Extraction Facility and while actions for management of nuclear materials, highly enriched uranium, surplus plutonium disposition, and sodium-bonded nuclear fuel would be ongoing.	TC
		5.1 Air Resources	
		Table 5-1 compares the cumulative concentrations of nonradiological air pollutants from the SRS to Federal and state regulatory standards. The listed values are the maximum modeled concentrations that could occur at ground level at the Site boundary. The data demonstrate that total estimated concentrations of nonradiological air pollutants from SRS would in all cases be below the regulatory standards at the Site boundary. The highest percentages of the regulatory standards are for sulfur dioxide concentrations for the shorter time interval (approximately 97 percent of standard for the 24-hour averaging time), for particulate matter of less than 10 microns (approximately 89 percent of standard for the 24-hour averaging time), and total suspended particulates (approximately 90 percent of standard on an annual basis). The remaining pollutant emissions would range from 1 to 69 percent of the applicable standards.	TC
EC	In addition, the cumulative impacts analysis includes the impacts from actions proposed in this SNF EIS. Risks to members of the public and site workers from radiological and nonradiological releases are based on operational impacts from the maximum impact alternative described in Section 4.1.2.		
EC	In addition, the cumulative impacts analysis accounts for other SRS operations. Most of the SRS baseline data are based on 1997 environmental report information (Arnett and Mamatey		
EC	1998), which are the most recent published data available.		TC
TC	Temporal limits were defined by examining the period of influence from both the proposed action and other Federal and non-Federal actions that have the potential for cumulative impacts. Actions for SNF management are expected to begin in 2000 in preparation for ultimate offsite disposal, possibly in a monitored geologic repository which probably will not be available until at least 2010. Final offsite shipments of	The majority of the impacts come from estimates of SRS baseline concentrations. It is unlikely that actual concentrations at ambient monitoring stations would be as high as that shown for the baseline values. The SRS baseline values are based on maximum potential emissions from the 1998 air emissions inventory and for all SRS sources, and observed concentrations from nearby ambient air monitoring stations.	TC

Table 5-1. Estimated maximum cumulative ground-level concentrations of nonradiological pollutants (micrograms per cubic meter) at SRS boundary.^{a,b}

Pollutant	Averaging time	SCDHEC ambient standard (µg/m ³)	SNF	SRS base-line (µg/m ³)	Other foreseeable planned SRS activities ^c (µg/m ³)	Cumulative concentration ^{d,e} (µg/m ³)	Percent of standard
Carbon monoxide	1 hour	40,000	9.760	10,000	36.63	10,046	25
	8 hours	10,000	1.31	6,900	5.15	6,906	69
Oxides of Nitrogen	Annual	100	3.36	26	4.38	33.7	34
Sulfur dioxide	3 hours	1,300	0.98	1,200	8.71	1,210	93
	24 hours	365	0.13	350	2.48	352.6	97
	Annual	80	0.02	34	0.17	34.2	43
Ozone ^f	1 hour	235	0.80	NA ^g	0.71	1.5	1
Lead	Max. quarter	1.5	NA	0.03	0.00	0.03	2
Particulate matter (≤10 microns aerodynamic diameter) ^f	24 hours	150	0.13	130	3.24	133.4	89
	Annual	50	0.02	25	0.13	25.2	50
Total suspended particulates (µg/m ³)	Annual	75	0.02	67	0.06	67.1	89

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TC

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a. DOE (1994; 1996; 1998; 1999b,c,d) and Hunter (1999) for baseline values.

b. Hydrochloric acid, formaldehyde, hexane, and nickel are not listed in Table 5-1 because operation of SNF or other foreseeable, planned SRS activities would not result in any change to the SRS baseline concentrations of these toxic pollutants.

c. Includes Highly Enriched Uranium, Tritium Extraction Facility, Management of Certain Plutonium Residues and Scrub Alloy Concentrations, Defense Waste Processing Facility, and Disposition of Surplus Plutonium, Sodium-Bonded Spent Nuclear Fuel, and components from throughout the DOE complex.

d. SCDHEC (1976).

e. Includes SNF concentrations.

f. New NAAQS for ozone (1 hr replaced by 8 hr standard = 0.08 ppm) and particulate matter ≤ 2.5 microns (24 hr standard = 65 µg/m³) and annual standard of 15 µg/m³ will become enforceable during the stated temporal range of the cumulative impacts analyses.

g. Not available.

DOE also evaluated the cumulative impacts of airborne radioactive releases in terms of dose to a maximally exposed individual at the SRS boundary. DOE included the impacts of Plant Vogtle (NRC 1996) in this cumulative total. The radiological emissions from the operation of the Chem-Nuclear low-level waste disposal facility just east of SRS are very low (SCDHEC 1992) and are not included.

Table 5-2 lists the results of this analysis, using 1997 emissions (1992 for Plant Vogtle) for the

SRS baseline. The cumulative dose to the maximally exposed member of the public would be 1×10^{-4} rem (or 0.1 millirem) per year, well below the regulatory standard of 10 millirem per year (40 CFR Part 61). Summing the doses to maximally exposed individual for the nine actions and baseline SRS operations listed in Table 5-2 is an extremely conservative approach because in order to get the calculated dose, the maximally exposed individual would have to occupy different physical locations at the same time, which is impossible.

TC

TC

Table 5-2. Estimated average annual cumulative radiological doses and resulting health effects to the maximally exposed offsite individual and population in the 50-mile radius from airborne releases.

	Activity	Offsite Population		
		Maximally exposed individual		50-mile population
		Dose (rem)	Probability of fatal cancer risk	Collective dose (person-rem) Excess latent cancer fatalities
TC	SRS Baseline ^a	5.0×10^{-5}	2.5×10^{-8}	2.2 1.1×10^{-3}
	Management of Spent Nuclear Fuel ^b	1.5×10^{-5}	7.5×10^{-9}	0.56 2.8×10^{-4}
	Surplus HEU Disposition ^c	2.5×10^{-6}	1.3×10^{-9}	0.16 8.0×10^{-5}
	Tritium Extraction Facility ^d	2.0×10^{-5}	1.0×10^{-8}	0.77 3.9×10^{-4}
	Surplus Plutonium Disposition ^e	7.4×10^{-6}	3.7×10^{-9}	1.8 9.0×10^{-4}
TC	Management of Plutonium Residues/Scrub Alloy ^f	5.7×10^{-7}	2.9×10^{-10}	6.2×10^{-3} 3.1×10^{-6}
	Defense Waste Processing Facility ^g	1.0×10^{-6}	5.0×10^{-10}	0.071 3.6×10^{-5}
L4-17	DOE complex miscellaneous components ^h	4.4×10^{-6}	2.2×10^{-9}	7.0×10^{-3} 3.5×10^{-6}
	Sodium-Bonded Spent Nuclear Fuel ⁱ	3.9×10^{-7}	2.0×10^{-10}	1.9×10^{-2} 9.5×10^{-6}
	Plant Vogtle ^j	5.4×10^{-7}	2.7×10^{-10}	0.042 2.1×10^{-5}
TC	Total	1.0×10^{-4}	5.1×10^{-8}	5.6 2.8×10^{-3}
TC	a. Arnett and Mamatey (1998) for 1997 data for MEI and population.			
	b. Maximum-impact alternative.			
	c. DOE (1996); HEU = highly enriched uranium.			
	d. DOE (1999b).			
	e. DOE (1999c).			
TC	f. DOE (1998).			
EC	g. DOE (1994).			
	h. Derive from impacts from conventional processing of Group A fuel.			
	i. DOE (1999d).			
	j. NRC (1996).			

Adding the population doses from current and projected activities at SRS, Plant Vogtle, and management of SNF could yield a total annual cumulative dose of 5.6 person-rem from airborne sources. The total annual cumulative dose translates into 2.8×10^{-3} latent cancer fatality for each year of exposure for the population living within a 50-mile (80-kilometer) radius of the SRS. For comparison, 143,863 deaths from cancer due to all causes would be likely in the same population over their lifetimes.

5.2 Water Resources

At present, a number of SRS facilities discharge treated wastewater to Upper Three Runs and its tributaries and Fourmile Branch via National Pollutant Discharge Elimination System

(NPDES)—permitted outfalls. These include the F and H Area Effluent Treatment Facility (ETF) and the M-Area Liquid Effluent Treatment Facility. As stated in Section 4.1.1.1, SNF operations are not expected to result in any discharges to groundwater. The only technology that would result in discharges of radioactive and nonradioactive effluents to surface water would be Conventional Processing. The major sources of liquid effluents from facilities associated with Conventional Processing would be process cooling water and steam condensate systems that could contain small quantities of radionuclides and chemicals. This process wastewater would be treated at ETF and then discharged to Upper Three Runs. Studies of water quality and biota downstream of the ETF outfall suggest that discharges from it have not degraded the

water quality of Upper Three Runs. Other potential sources of contaminants into Upper Three Runs during the SNF management period include the accelerator production of tritium, the tritium extraction facility, environmental restoration, and decontamination and decommissioning activities, as well as modifications to existing SRS facilities. Discharges associated with the accelerator production of tritium and tritium extraction facility activities would not add significant amounts of nonradiological contaminants to Upper three Runs. The amount of discharge associated with environmental restoration and decontamination and decommissioning activities would vary based on the level of activity. All the potential activities that could result in wastewater discharges would be required to comply with the NPDES permit limits that ensure protection of water quality. Studies of water quality and biota in Upper Three Runs suggest that discharges from facilities outfalls have not degraded the stream (Halverson et al. 1997).

Table 5-3 summarizes the estimated cumulative radiological doses from waterborne sources to human receptors downstream from SRS. Liquid effluents would be released to SRS streams that are tributaries of the Savannah River could contain small quantities of radionuclides. The exposure pathways considered in this analysis included drinking water, fish ingestion, shoreline exposure, swimming, and boating. The estimated cumulative dose to the maximally exposed member of the public from liquid releases would be 2.4×10^{-4} rem (or 0.24 millirem) per year, well below the regulatory standard of 4 millirem per year (40 CFR Part 141). Adding the population doses associated with current and projected SRS activities would yield a cumulative annual dose of 2.6 person-rem from liquid sources. This translates into 0.0013 latent cancer fatality for each year of exposure of the population living within a 50-mile (80-kilometer) radius of the SRS. For comparison,

15,300 deaths from cancer due to all causes would be likely in the population of 70,000 downstream residents over their lifetimes.

5.3 Public and Worker Health

Table 5-4 summarizes the cumulative radiological health effects of routine SRS operations, proposed DOE actions, and non-Federal nuclear facility operations (Plant Vogtle Electric Generating Facility). Impacts resulting from proposed DOE actions are described in the EISs listed previously in this chapter. In addition to estimated radiological doses to the hypothetical maximally exposed offsite individual, the offsite population, and involved workers, Table 5-4 also lists the potential number of latent cancer fatalities for the public and workers due to exposure to radiation. The radiation dose to the maximally exposed offsite individual from air and liquid pathways would be 3.4×10^{-4} rem (0.34 mrem) per year, which is well below the applicable DOE regulatory limits (10 mrem per year from the air pathway, 4 mrem per year from the liquid pathway, and 100 mrem per year for all pathways). The total annual population dose for current and projected activities of 8.2 person-rem translates into 0.004 latent cancer fatality for each year of exposure for the population living within a 50-mile (80-kilometer) radius of the SRS. As stated in Section 5.1, for comparison, 143,863 deaths from cancer due to all causes would be likely in the same population over their lifetimes.

The annual radiation dose to the involved worker population would be 859 person-rem. In addition, doses to individual workers would be kept below the regulatory limit of 5,000 mrem per year (10 CFR 835). Furthermore, as low as reasonably achievable principles would be exercised to maintain individual worker doses below the DOE Administrative Control Level of 2,000 mrem per year.

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Table 5-3. Estimated average annual cumulative radiological doses and resulting health effects to offsite population in the 50-mile radius from aqueous releases.

Activity	Offsite Population			
	Maximally exposed individual		50-mile population	
	Dose (rem)	Probability of fatal cancer risk	Collective dose (person-rem)	Excess latent cancer fatalities
SRS Baseline ^a	1.3×10^{-4}	6.5×10^{-8}	2.4	1.1×10^{-3}
Management of Spent Nuclear Fuel ^b	5.7×10^{-5}	2.9×10^{-8}	0.19	9.5×10^{-5}
Surplus HEU Disposition ^c	(d)	(d)	(d)	(d)
Tritium Extraction Facility ^e	(d)	(d)	(d)	(d)
Defense Waste Processing Facility ^f	(d)	(d)	(d)	(d)
Surplus Plutonium Disposition ^g	(d)	(d)	(d)	(d)
Management Plutonium Residues/Scrub Alloy ^h	(d)	(d)	(d)	(d)
DOE complex miscellaneous components ⁱ	4.2×10^{-8}	2.1×10^{-11}	2.4×10^{-4}	1.2×10^{-7}
Sodium-Bonded Spent Nuclear Fuel ^j	1.2×10^{-7}	6.0×10^{-11}	6.8×10^{-4}	3.4×10^{-7}
Plant Vogtle ^k	5.4×10^{-5}	2.7×10^{-8}	2.5×10^{-3}	1.3×10^{-6}
Total	2.4×10^{-4}	1.2×10^{-7}	2.6	1.3×10^{-3}

a. Arnett and Mamatey (1998) for 1997 data for MEI and population. Worker dose is based on 1997 data (WSRC 1998).

b. Maximum-impact alternative.

c. DOE (1996); HEU = highly enriched uranium.

d. Less than minimum reportable levels.

e. DOE (1999b).

f. DOE (1994).

g. DOE (1999c).

h. DOE (1998).

i. Derived from impacts from conventional processing.

j. DOE (1999d).

k. NRC (1996).

5.4 Waste Generation

As stated in Section 4.1.1.4, high-level waste, transuranic waste, and low-level waste would be generated from SNF management activities. Smaller amounts of mixed and hazardous waste would also be generated from SNF processing activities. The largest volume of high-level and transuranic waste would be generated with the Conventional Processing alternative. However, as stated in Section 4.1.1.4, the projected high-level waste and transuranic waste generation

rates would not require additional treatment and storage capacities beyond the current and planned SRS capacities. In general, the waste generation rate varies with each phase of SNF handling and the type of fuel group. The total radioactive/hazardous waste volume associated with SNF activities could range from 20,700 cubic meters (27,076 cubic yards) for the minimum impact option to 154,967 cubic meters (202,681 cubic yards) for the maximum impact (conventional processing) option.

Table 5-4. Estimated average annual cumulative radiological doses and resulting health effects to offsite population and facility workers.

	Maximally exposed individual					Offsite population ^a				Workers	
	Activity	Dose from airborne releases (rem)	Dose from liquid releases (rem)	Total dose (rem)	Probability of fatal cancer risk	Collective dose from airborne releases (person-rem)	Collective dose from liquid releases (person-rem)	Total collective dose (person-rem)	Excess latent cancer fatalities	Collective dose	Excess latent cancer fatalities
TC	SRS Baseline ^b	5.0×10 ⁻⁵	1.3×10 ⁻⁴	1.8×10 ⁻⁴	9.0×10 ⁻⁸	2.2	2.4	4.6	2.3×10 ⁻³	165	0.066
	Management of Spent Nuclear Fuel ^c	1.5×10 ⁻⁵	5.7×10 ⁻⁵	7.2×10 ⁻⁵	3.6×10 ⁻⁸	0.56	0.19	0.75	3.8×10 ⁻⁴	55	0.022
	Surplus HEU Disposition ^d	2.5×10 ⁻⁶	(e)	2.5×10 ⁻⁶	1.3×10 ⁻⁸	0.16	(e)	0.16	8.0×10 ⁻⁵	11	4.4×10 ⁻³
	Tritium Extraction Facility ^f	2.0×10 ⁻⁵	(e)	2.0×10 ⁻⁵	1.0×10 ⁻⁸	0.77	(e)	0.77	3.9×10 ⁻⁴	4	1.6×10 ⁻³
	Defense Waste Processing Facility ^g	1.0×10 ⁻⁶	(e)	1.0×10 ⁻⁶	5.0×10 ⁻¹⁰	0.071	(e)	0.071	3.6×10 ⁻⁵	120	0.048
L4-17	Surplus Plutonium Disposition ^h	7.4×10 ⁻⁶	(e)	7.4×10 ⁻⁶	3.7×10 ⁻⁹	1.8	(e)	1.8	9.0×10 ⁻⁴	456	0.18
	Management Plutonium Residues/Scrub Alloy ⁱ	5.7×10 ⁻⁷	(e)	5.7×10 ⁻⁷	2.9×10 ⁻¹⁰	6.2×10 ⁻³	(e)	6.2×10 ⁻³	3.1×10 ⁻⁶	7.6	3×10 ⁻³
	DOE complex miscellaneous components ^j	4.4×10 ⁻⁶	4.2×10 ⁻⁸	4.4×10 ⁻⁶	2.2×10 ⁻⁹	7.0×10 ⁻³	2.4×10 ⁻⁴	7.2×10 ⁻³	3.6×10 ⁻⁶	2	0.001
	Sodium-Bonded Spent Nuclear Fuel ^k	3.9×10 ⁻⁷	1.2×10 ⁻⁷	5.1×10 ⁻⁷	2.6×10 ⁻¹⁰	1.9×10 ⁻²	6.8×10 ⁻⁴	2.0×10 ⁻²	9.8×10 ⁻⁶	38	0.015
	Plant Vogtle ^l	5.4×10 ⁻⁷	5.4×10 ⁻⁵	5.5×10 ⁻⁵	2.7×10 ⁻⁸	0.042	2.5×10 ⁻³	0.045	2.2×10 ⁻⁵	NA	NA
TC	Total	1.0×10 ⁻⁴	2.4×10 ⁻⁴	3.4×10 ⁻⁴	1.7×10 ⁻⁷	5.6	2.6	8.2	4.1×10 ⁻³	859	0.34

N/A = not available

a. A collective dose to the 50-mile (80-kilometer) population for atmospheric releases and to the downstream users of the Savannah River for aqueous releases.

b. Arnett and Mamatey (1998) for 1997 data for MEI and population. Worker dose is based on 1997 data (WSRC 1998).

c. Maximum-impacts alternative.

d. DOE (1996); HEU = highly enriched uranium.

e. Less than minimum reportable levels.

f. DOE (1999b).

g. DOE (1994).

h. DOE (1999c).

i. DOE (1998).

j. Derived from impacts from conventional processing of Group A fuel.

k. DOE (1999d).

NRC (1996).

Table 5-5 lists cumulative volumes of high-level, low-level, transuranic, and hazardous and mixed wastes that SRS would generate. The table includes data from the SRS 30-year expected waste forecast (WSRC 1994). The 30-year expected waste forecast is based on operations, environmental restoration, and decontamination and decommissioning waste forecasts from existing generators and the following assumptions: secondary waste from the Defense Waste Processing Facility, In-Tank Precipitation, and Extended Sludge Processing operations are addressed in the DWPF EIS; high-level waste volumes are based on the selected option for the F-Canyon Plutonium Solutions EIS; some investigation-derived wastes are handled as hazardous waste per Resource Conservation and Recovery Act (RCRA) regulations; purge water from well samplings is handled as hazardous waste; and the continued receipt of small amounts of low-level waste from other DOE facilities and nuclear naval operations. The estimated quantity of radioactive/hazardous waste from operations in this forecast during the next 30 years would be 142,666 cubic meters. In addition, radioactive/hazardous waste associated with environmental restoration and decontamination and decommissioning activities would have a 30-year expected forecast of 67,808 cubic meters (Halverson 1999). Waste generated from the conventional processing option would add a total of 154,970 cubic meters. During this same time period, other reasonably foreseeable activities that were not included in the 30-year forecast would add an additional 192,915 cubic. The major contributor to the other waste volumes would be from weapons components from various DOE sites that could be processed in SRS canyons. Therefore, the potential cumulative amount of waste generated from SRS activities during the period of interest would be 558,359 cubic meters. It is important to note that the quantities of waste generated are not equivalent to the amounts that will require disposal. As discussed in Section 4.1.1.4 for example, high-level waste is evaporated and concentrated to a smaller volume for final disposal. Combustible low-level waste is volume

reduced on site in the Consolidated Incineration Facility.

The Three Rivers Solid Waste Authority Regional Waste Management Center at the Savannah River Site accepts non-hazardous and non-radioactive solid wastes from SRS and eight surrounding South Carolina counties. This municipal solid waste landfill provides state of the art Subtitle D (non-hazardous) facilities for landfilling solid wastes while reducing the environmental consequences associated with construction and operation of multiple county-level facilities (DOE 1995b). It was designed to accommodate combined SRS and county solid waste disposal needs for at least 20 years, with a projected maximum operational life of 45 to 60 years (DOE 1995b). The landfill is designed to handle an average of 1,000 tons per day and a maximum of 2,000 tons per day of municipal solid wastes. The SRS and eight cooperating counties had a combined generation rate of 900 tons per day in 1995. The Three Rivers Solid Waste Authority Regional Waste Management Center opened in mid-1998.

The SNF management activities and other planned SRS activities would not generate larger volumes of radioactive, hazardous, or solid wastes beyond current and projected capacities of SRS waste storage and/or management facilities.

5.5 Utilities and Energy

Table 5-6 lists the cumulative consumption of electricity from activities at SRS. The values are based on annual consumption estimates. Among the SNF management technologies, Conventional Processing would place the largest annual demand on electricity and water resources. The SNF management values are based on the maximum impact analysis (Section 4.1.1.5).

The overall SRS activities occurring concurrently with SNF management activities would not place an unreasonable demand on electricity resources.

Table 5-5. Estimated cumulative waste generation from SRS concurrent activities (cubic meters).^{a,b,c}

Waste Type	SNF Management ^a	SRS Operations ^{b,c}	ER/D&D ^{b,c,d}	Other Waste Volume ^e	Total
High-level	11,000	14,129	0	69,552	94,681
Low-level	140,000	118,669	61,630	110,102	430,401
Hazardous/mixed	270	3,856	6,178	4,441	14,745
Transuranic	3,700	6,012	0	8,820	18,532
Total	154,970	142,666	67,808	192,915	558,359

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a. Maximum-impact alternative.

b. Halverson (1999).

c. Based on a total 30-year expected waste generation forecast, which includes previously generated waste.

d. ER/D&D = environmental restoration/decontamination & decommissioning.

e. Life-cycle waste associated with reasonably foreseeable future activities such as TEF, plutonium residues, surplus plutonium disposition, highly-enriched uranium, commercial light water reactor waste, sodium-bonded spent nuclear fuel, and weapons components that could be processed in SRS canyons. Impacts for the last group is based on conventional processing impacts for SNF Fuel Group A.

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EC**Table 5-6.** Estimated average annual cumulative utility consumption.

Activity	Electricity (megawatt-hours)	Water usage (liters)
SRS baseline ^a	4.11×10^5	1.70×10^{10}
SNF management ^b	1.58×10^4	2.11×10^8
Other SRS foreseeable activities	1.51×10^5	6.73×10^8
Total	5.77×10^5	1.79×10^{10}

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a. Halverson (1999) for electricity usage and Arnett and Mamatey (1996) for water usage.

b. Based on the maximum impact alternative.

c. Includes utility consumption associated with reasonable foreseeable future actions such as tritium extraction, facility, plutonium residues, surplus plutonium disposition, highly-enriched uranium, sodium-bonded spent nuclear fuel, and weapons components that could be processed at SRS canyons. Impacts for last group are based on conventional processing impacts of spent nuclear fuel "Group A." See EISs referenced at end of chapter. Sodium-bonded spent nuclear fuel electricity usage based on "Group A" conventional processing; water usage from EIS.

DOE has also evaluated the SRS water needs during the SNF management activities period. At present, the SRS rate of groundwater withdrawal is estimated to be up to 17 billion liters annually. The estimated amount of groundwater needed for SNF management activities from 1998 to 2035 is 211 million liters per year, depending on the management option chosen. Operation of other foreseeable activities would require approximately 673 million liters of groundwater per year. Thus, sitewide groundwater withdrawals would increase minimally over the projected SNF management period.

Surface water usage during the SNF management period is not projected to approach capacity levels.

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5.6 Socioeconomic Impacts

Cumulative regional economic and population changes from construction and operation of the Transfer and Storage Facility or the Transfer, Storage and Treatment Facility consider the impacts of other coincident economic development projects such as DOE's Accelerator for the Pro-

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duction of Tritium, Bridgestone-Firestone, and Hankook Synthetics.

TC | Bridgestone-Firestone is building a \$435 million tire manufacturing plant in Aiken County that will employ 800 workers. The Bridgestone-Firestone project is expected to complete construction and be in operation by the year 2000. Thus, this project should not impact the construction workforce for the Transfer and Storage Facility or Transfer, Storage and Treatment Facility which are not scheduled to be constructed until after the year 2000. Competition for construction workers should not overlap.

Construction of the Transfer and Storage Facility or the transfer and storage phase of the

Transfer, Storage and Treatment Facility would begin sometime after the year 2000, employ 500 workers (375 construction and 125 professional), and require 2 years to complete. The treatment phase would begin construction at the completion of the transfer and storage phases and also could employ as many as 500 workers and take as long as 2 years to complete. No additional workers would be required during operations since existing SRS employees would assume those positions.

There would be no significant cumulative socioeconomic impacts from construction or operation of the Transfer and Storage Facility or the Transfer, Storage and Treatment Facility.

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References

- Arnett, M. W., and A. R. Mamatey, 1996, *Savannah River Site Environmental Report for 1995*, WSRC-TR-96-0075, Westinghouse Savannah River Company, Savannah River Site Aiken, South Carolina. TC
- Arnett, M. W., and A. R. Mamatey, 1998, *Savannah River Site Environmental Data for 1997*, WSRC-TR-97-00322, Westinghouse Savannah River Company, Aiken, South Carolina. TC
- CEQ (Council on Environmental Quality), 1997, *Considering Cumulative Effects Under the National Environmental Policy Act*, Executive Office of the President, Washington, D.C.
- DOE (U.S. Department of Energy), 1994, *Final Defense Waste Processing Facility Supplemental Environmental Impact Statement*, DOE/EIS 0082-S, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1995a, *Final Environmental Impact Statement - Interim Management of Nuclear Materials*, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1995b, *Environmental Assessment for the Construction and Operation of the Three Rivers Solid Waste Authority Regional Waste Management Center at the Savannah River Site*, DOE/EA-1079, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1996, *Disposition of Surplus Highly Enriched Uranium Final Environmental Impact*, DOE/EIS-0240, Office of Fissile Materials Disposition, Washington, D.C. EC
- DOE (U.S. Department of Energy), 1997, *Environmental Assessment for the Tritium Facility Modernization and Consolidation Project at the Savannah River Site*, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1998, *Final Environmental Impact Statement on Management of Certain Plutonium Residues and Scrub Alloy at the Rocky Flats Environmental Technology Site*, DOE/EIS-0277F, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1999a, *Final Environmental Impact Statement Accelerator Production of Tritium at Savannah River Site*, DOE/EIS-0270, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1999b, *Final Environmental Impact Statement for the Construction and Operation of a Tritium Extraction Facility at the Savannah River Site*, DOE/EIS-0271, Savannah River Operations Office, Aiken, South Carolina, March.
- DOE (U.S. Department of Energy), 1999c, *Surplus Plutonium Disposition Final Environmental Impact Statement*, DOE/EIS-0283, Office of Fissile Materials Disposition, Washington, D.C. EC
- DOE (U.S. Department of Energy), 1999d, *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*, DOE/EIS-0306D, Office of Nuclear Energy, Science and Technology, Washington, D.C.
- Halverson, N. V., L. D. Wike, K. K. Patterson, J. A. Bowers, A. L. Bryan, K. F. Chen, C. L. Cummins, B. R. del Carmen, K. L. Dixon, D. L. Dunn, G. P. Friday, J. E. Irwin, R. K. Kolka, H. E. Mackey, Jr.,

- J. J. Mayer, E. A. Nelson, M. H. Paller, V. A. Rogers, W. L. Specht, H. M. Westbury, and E. W. Wilde, 1997, *SRS Ecology Environmental Information Document*, WSRC-TR-97-0223, Westinghouse Savannah River Company, Aiken, South Carolina.
- Hess, M. L., 1995, Westinghouse Savannah River Company, Aiken, South Carolina, Interoffice Memorandum to H. L. Pope, U.S. Department of Energy, Savannah River Operations Office, Aiken, South Carolina, "WSRC Data Transmittal - Summary of Changes Between Draft and Final WMEIS Forecasts and Alternatives," ESH-NEP-95-0084, May 15.
- Hunter, C. H., 1999, Westinghouse Savannah River Company, Aiken, South Carolina, Interoffice memorandum to C. B. Shedrow, "Non-Radiological Air Quality Modeling for the High-Level Waste Tank Closure Environmental Impact Statement (EIS)," SRT-NTS-990067, March 26.
- NRC (U.S. Nuclear Regulatory Commission), 1996, *Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites in 1992*, NUREG/CR 2850, Vol. 14, Washington, D.C.
- SCDHEC (South Carolina Department of Health and Environmental Control), 1976, *South Carolina Code of Laws*, SCDHEC Regulation, "Air Pollution Control Regulations and Standards," Columbia, South Carolina.
- SCDHEC (South Carolina Department of Health and Environmental Control), 1992, *South Carolina Nuclear Facility Monitoring - Annual Report 1992*, Columbia, South Carolina.
- SCDHEC (South Carolina Department of Health and Environmental Control), 1995, *South Carolina Nuclear Facility Monitoring - Annual Report 1995*, Columbia, South Carolina.
- WSRC (Westinghouse Savannah River Company), 1994, *Thirty-Year Solid Waste Generation Forecast for Facilities at SRS*, Revision 3, WSRC-RP-94-532, Savannah River Site, Aiken, South Carolina.
- WSRC (Westinghouse Savannah River Company), 1998, *Savannah River Site Radiological Performance*, 4th Quarter 1997, ESH-SHP-98-0007, Savannah River Site, Aiken, South Carolina.

CHAPTER 6. RESOURCE COMMITMENTS

6.1 Introduction

Chapter 6.0 describes the unavoidable adverse impacts, short-term uses of environmental resources versus long-term productivity, and irreversible or irretrievable commitments of resources associated with safely managing spent nuclear fuel (SNF) at the Savannah River Site (SRS) for the period 1998 to 2035. This chapter also includes discussions about U.S. Department of Energy (DOE) waste minimization, pollution prevention, and energy conservation programs as they would relate to implementation of the proposed action.

6.2 Unavoidable Adverse Impacts

Implementing any of the alternatives considered in this environmental impact statement (EIS) for the management of SNF at SRS would result in minimal unavoidable adverse impacts to the human environment. Construction and operation of a Transfer and Storage Facility to implement the New Packaging Technology or the construction and operation of a Transfer, Storage, and Treatment Facility to implement the New Processing Technology would result in negligible adverse impacts to geologic resources, groundwater, traffic, and cultural resources as described in Chapter 4. All construction activities would occur within the boundary of a reactor or a chemical separations area in an already-developed industrial complex and would require approximately 15 acres.

Potential adverse impacts from construction could occur to surface water resources. However, as part of the required sediment and erosion control plan, storm water management and sediment control measures would minimize runoff from the construction site and potential discharges of silts, solids, and other contaminants to surface-water streams. There would be minimal adverse impacts to air resources from construction activities. Concentrations of pollutants emitted during construction activities

would be at least an order of magnitude less than the South Carolina ambient air quality standards concentrations. Likewise, there would be minimal adverse impacts to the ecological resources of the area, primarily due to construction-related noises. Although noise levels would be relatively low outside the immediate area of construction, the combination of construction noise and human activity probably would displace small numbers of animals. These adverse impacts would be small, temporary (24 months or less), and localized. Construction would not disturb any threatened or endangered species, would not degrade any critical or sensitive habitat, and would not affect any jurisdictional wetlands.

Renovating an existing facility for the Transfer, Storage, and Treatment Facility could result in additional low-level waste generation, which could be considered a potential adverse impact. Renovation would require decontamination and removal of components and systems and subsequent construction inside a building, such as a reactor building. Adverse impacts would include the generation of approximately 480 m³ of low-level radioactive waste. This waste volume would have minimal impact on the Site's overall waste management capacity. Eventual decontamination and decommissioning (D&D) of any facility (either new and dedicated to SNF management or renovated to accommodate SNF management) used for the management of SNF would result in the generation of radioactive waste. Impacts of these D&D activities would be evaluated in subsequent National Environmental Policy Act (NEPA) actions.

Unavoidable construction worker radiation exposures would result from renovating an existing reactor facility to become the Transfer, Storage, and Treatment Facility. These occupational exposures (32 person-rem in a population of 54 construction workers) would be well below regulatory limits.

6.3 Relationship between Local Short-Term Uses of the Environment and the Maintenance and Enhancement of Long-Term Productivity

The proposed locations for any new facility are all within developed industrial landscapes. Each of the proposed sites would encompass approximately 15 acres. The existing infrastructure (roads; power-, steam-, and waterlines; wastewater treatment facilities, etc.) within each of the areas is sufficient to support the proposed facilities.

Regardless of location, after the operational life of the project, DOE could decontaminate and decommission (D&D) the facility in accordance with applicable regulatory requirements and restore the area to a brown-field site that would be available for other industrial use. Appropriate NEPA reviews would be conducted prior to the initiation of any D&D action. In all likelihood, none of the sites would be restored to a natural terrestrial habitat.

The project-related uses of environmental resources for the duration of any of the proposed alternatives are characterized below.

- Over the life of the SNF management alternatives, groundwater would be used to meet sanitary and process water needs. After use and treatment, this water would be discharged into surface water streams. Depending on the site chosen and the technology implemented, over the short-term, the resulting increases in pollutant loadings would take advantage of the natural assimilative capacity of the receiving stream(s). However, these incremental pollutant loadings should not adversely affect either short- or long-term productivity of the aquatic ecosystem. These impacts would be assessed during the regulatory permitting process once an alternative has been selected.

- Regardless of location, air emissions associated with implementation of any of the technologies would add small amounts of radiological and nonradiological constituents to the air of the region. During the project's life, these emissions would result in an additional loading and exposure but would not impact SRS compliance with air quality or radiation exposure standards. There would be no significant residual environmental affects to long-term environmental productivity.
- The management and disposal of sanitary solid waste and non-recyclable radiological waste over the project's life would require energy and space at SRS treatment, storage, or disposal facilities (e.g., Three Rivers Sanitary Landfill, E-Area Vaults, Consolidated Incineration Facility). The land required to meet the solid waste needs would require a long-term commitment of terrestrial resources. Upon the facilities' closures, DOE could D&D them and restore them to brown field sites which could be available for future commercial or industrial development.
- Regardless of location, increased employment, expenditures, and tax revenues generated during the implementation of any of the alternatives would directly benefit the local, regional, and state economies over the short-term. Long-term economic productivity could be facilitated by local governments investing project-generated tax revenues into infrastructure and other required services.

6.4 Irreversible and Irrecoverable Resource Commitments

Resources that would be irreversibly and irretrievably committed during the implementation of SNF management alternatives include those that cannot be recovered or recycled and those that are consumed or reduced to unrecoverable

forms. The commitment of capital, energy, labor, and material during the implementation of SNF management alternatives would generally be irreversible.

Energy expended would be in the form of fuel for equipment and vehicles, electricity for facility operations, and human labor. Construction would generate nonrecyclable materials such as sanitary solid waste and construction debris. Operation of any proposed facility would generate nonrecyclable waste streams such as radiological and nonradiological solid wastes and some process wastewaters. However, certain materials (e.g., copper, stainless steel) used during construction and operation of the proposed facility could be recycled when the facility was D&Ded. Some construction materials, particularly from existing facilities (e.g., Receiving Basin for Offsite Fuel, L-Reactor Disassembly Area, F- and H-Separation Facilities) would not be salvageable due to radioactive contamination. Table 6-1 lists estimated requirements for concrete and steel for any new facility.

Table 6-2 lists the major materials that would be consumed as a result of process operations, primarily chemicals and other commercial products. Table 2-4 lists the corresponding management technologies that would use the facilities.

The implementation of the SNF management alternatives considered in this EIS, including the No-Action Alternative, would require water, electricity, steam, and diesel fuel. Tables 4.1-15 through 4.1-18 list estimated amounts of these resources that would be consumed during the period of analysis; Section 4.1.1.5 describes the uses. Water would be obtained from onsite groundwater sources and steam from existing onsite sources. Electricity and diesel fuel would be purchased from commercial sources. These commodities are readily available and the amounts required would not have an appreciable impact on available supplies or capacities. From a materials and energy resource commitment perspective, Conventional Processing and the Electrometallurgical Treatment Technology

option would recover low enriched uranium, which is useable as commercial reactor fuel. None of the other alternatives would recover this resource.

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6.5 Waste Minimization, Pollution Prevention, and Energy Conservation

6.5.1 WASTE MINIMIZATION AND POLLUTION PREVENTION

DOE has implemented an aggressive waste minimization and pollution prevention program at SRS at the sitewide level and for individual organizations and projects. As a result, significant reductions have been achieved in the amounts of wastes discharged into the environment and sent to landfills, resulting in significant cost savings.

To implement a waste minimization and pollution prevention program at the SNF management facilities, DOE would characterize waste streams and identify opportunities for reducing or eliminating them. Emphasis would be placed on minimizing the largest waste stream, low-level waste, through source reduction and recycling. Selected waste minimization practices could include:

- Process design changes to eliminate the potential for spills and to minimize contamination areas
- Decontamination of equipment to facilitate reuse
- Recycling metals and other usable materials, especially during the construction phase of the project
- Preventive maintenance to extend process equipment life
- Modular equipment designs to isolate potential failure elements to avoid changing out entire units.

L1-5

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Table 6-1. Estimated requirements for concrete and steel for stand-alone facilities.

Facility	Concrete (cubic yards) ^a	Steel (tons) ^b
Transfer and Storage Facility (including dry storage vaults)	11,000	600
Transfer, Storage, and Treatment Facility (construction of new facility)	20,000	1,800

a. To convert cubic yards to cubic meters, multiply by 0.764.

b. To convert tons to metric tons, multiply by 0.907.

Table 6-2. Major chemicals and other materials required for spent nuclear fuel management facilities.

Facility	Major material requirements (operation)
Receiving Basin for Offsite Fuel	Water treatment filters, deionizer resins
L-Reactor Disassembly Basin	Water treatment filters, deionizer resins
F or H Canyon	Nitric acid, gelatin, tributyl phosphate, n-paraffin, depleted uranium
Transfer and Storage Facility	Nuclear poison, helium, neutron absorbers, stainless steel (canisters), water treatment filters and deionizer resins (if receipt basin is used)
Melt and Dilute Treatment Facility	Depleted uranium, neutron poison, helium, stainless steel (canisters), glass formers (glass or ceramic frit, silicon dioxide)
Mechanical Dilution Treatment Facility	Depleted uranium, nuclear poison (e.g., borated steel), helium, stainless steel (canisters)
Vitrification Facility	Depleted uranium, glass or ceramic formers (e.g., silicon oxide), stainless steel (canisters), offgas treatment materials (filters, chemicals)
• Dissolve and Vitrify • Glass Material Oxidation and Dissolution System • Plasma Arc	• Nitric acid, boric acid • Boron oxide, lead dioxide (mostly reused in the process), carbon • Offgas treatment materials (filters, chemicals)
Electrometallurgical Treatment Facility	Depleted uranium; glass; silicon; lithium fluoride, potassium fluoride, and uranium fluoride electrolytes; aluminosilicate filters; waste separation materials (ion exchange media or chemical reduction/oxide precipitation chemicals)

L1-5

- Use of non-toxic or less toxic materials to prevent pollution and minimize hazardous and mixed waste streams

During construction, DOE would implement actions to control surface water runoff and construction debris and to prevent infiltration of contaminants into groundwater. The construc-

tion contractor would be selected, in part, based on prior pollution prevention practices.

6.5.2 ENERGY CONSERVATION

SRS has an active energy conservation and management program. Since the mid-1990s more than 40 onsite administrative buildings

have undergone energy efficiency upgrades. Representative actions include the installation of energy-efficient light fixtures, the use of occupancy sensors in rooms, use of diode light sticks in exit signs, and the installation of insulating blankets around hot water heaters. Regardless

of location, the incorporation of these types of energy-efficient technologies into facility design, along with the implementation of process efficiencies and waste minimization concepts, would facilitate energy conservation by any of the SNF management alternatives.

CHAPTER 7. APPLICABLE LAWS, REGULATIONS, AND OTHER REQUIREMENTS

This chapter identifies and summarizes the major laws, regulations, Executive Orders, and Department of Energy (DOE) Orders that could apply to the management of spent nuclear fuel (SNF) at the Savannah River Site (SRS). Permits or licenses could be required under some of these laws and regulations. However, DOE would determine the specific requirements for permits or licenses, which would depend greatly on the chosen alternative, after consultation with the appropriate regulating agencies.

Section 7.1 discusses the major Federal and State of South Carolina statutes and regulations that impose environmental protection requirements on DOE and which require DOE to obtain a permit prior to construction and operation of spent nuclear fuel facilities. Each of the applicable regulations establishes how potential releases of pollutants and radioactive materials are to be controlled or monitored and include requirements for the issuance of permits for new operations or new emission sources. In addition to environmental permit requirements, the statutes may require consultations with various authorities to determine if an action (such as construction and operation of a facility) requires a permit or the implementation of protective or mitigative measures. Sections 7.1.1 and 7.1.2 discuss the environmental permitting process and lists the environmental permits and consultations (see Table 7-1) applicable to construction and operation of the spent nuclear fuel facilities.

Sections 7.2 and 7.3 address the major Federal regulations and Executive Orders, respectively, which address issues such as protection of public health and the environment, worker safety, and emergency planning. The Executive Orders clarify issues of national policy and set guidelines under which Federal agencies must act.

DOE implements its responsibilities for protection of public health, safety, and the environment through a series of Departmental Orders

(see Section 7.4) that are mandatory for operating contractors of DOE-owned facilities.

7.1 Statutes and Regulations Requiring Permits or Consultations

Environmental regulations require that the owner or operator of a facility obtain permits for the construction and operation of new (water and air) emissions sources, and for new domestic drinking water systems. To obtain these permits, the facility operator must apply to the appropriate government agency for a discharge permit for discharges of wastewater to the waters of the state and submit construction plans and specifications for the new emission sources, including new air sources. The environmental permits contain specific conditions with which the permittee must comply during construction and operation of a new emission source, describe pollution abatement and prevention methods to be utilized for reduction of pollutants, and contain emissions limits for pollutants which will be emitted from the facility. Section 7.1.1 discusses the environmental statutes and regulations under which DOE will be required to obtain permits. Table 7-1 lists the permits.

7.1.1 ENVIRONMENTAL PROTECTION PERMITS

Clean Air Act, as amended, (42 USC 7401 et seq.), (40 CFR Parts 50-99); South Carolina Pollution Control Act [Section 48-1-30 et seq., South Carolina Department of Health and Environmental Control (SCDHEC) Regulation 61-62]

The Clean Air Act, as amended, is intended to "protect and enhance the quality of the Nation's air resources so as to promote the public health and welfare and the productive capacity of its

*Applicable Laws, Regulations, and Other Regulations***Table 7-1. Environmental permits and consultations required by law.**

Activity/Topic	Law	Requirements	Agency
Site Preparation	Federal Clean Water Act (Section 404)	Stormwater Pollution Prevention Plan for Industrial Activity	SCDHEC ^a
Wastewater Discharges	Federal Clean Water Act S.C. Pollution Control Act	Stormwater Pollution Prevention/Erosion Control Plan for construction activity NPDES Permit(s) for Process Wastewater Discharges Process Wastewater Treatment Systems Construction and Operation Permits (if applicable) Sanitary Waste Water Pumping Station Tie-in Construction Permit; Permit to Operate	SCDHEC WSRC/EPD ^b SCDHEC SCDHEC SCDHEC WSRC/EPD
Air	Clean Air Act - NESHA ^c	Rad Emissions - Permit to construct new emission source (if needed) Air Construction and Operation permits - as required (e.g., Fire Water Pumps; Diesel Generators) General source - Stacks, Vents, Concrete batch plant Air Permit - Prevention of Significant Deterioration (PSD)	EPA ^c SCDHEC SCDHEC SCDHEC
Domestic Water	Safe Drinking Water Act	Construction and operation permits for line to domestic water system	WSRC/EPD SCDHEC
Endangered Species	Endangered Species Act	Consultation	U.S. Fish and Wildlife Service; National Marine Fisheries Service
Migratory Birds	Migratory Bird Treaty Act	Consultation	U.S. Fish and Wildlife Service

a. South Carolina Department of Health and Environmental Control.

b. WSRC/EPD Westinghouse Savannah River Company Environmental Protection Department.

c. Environmental Protection Agency.

population." Section 118 of the Clean Air Act, as amended, requires each Federal agency, such as DOE, with jurisdiction over any property or facility that might result in the discharge of air pollutants, to comply with "all Federal, State, interstate, and local requirements" with regard to the control and abatement of air pollution.

The Act requires the U.S. Environmental Protection Agency (EPA) to establish National Ambient Air Quality Standards as necessary to protect public health, with an adequate margin of safety, from any known or anticipated adverse effects of a regulated pollutant (42 USC 7409). The Act also requires the establishment of national standards of performance for new or modified stationary sources of atmospheric pollutants (42 USC 7411) and requires specific emission increases to be evaluated so as to prevent a significant deterioration in air quality (42 USC 7470). Hazardous air pollutants, including radionuclides, are regulated separately (42 USC 7412). Air emissions are regulated by the EPA in 40 CFR Parts 50 through 99. In particular, radionuclide emissions are regulated under the National Emission Standard for Hazardous Air Pollutants Program (NESHAP) (see 40 CFR Part 61).

EPA has overall authority for the Clean Air Act; however, it delegates primary authority to states which have an established air pollution control program approved by EPA. In South Carolina, EPA has retained authority over radionuclide emissions (40 CFR Part 61) and has delegated to SCDHEC the responsibility for the rest of the regulated pollutants under the authority of the South Carolina Pollution Control Act (48-1-10 et. seq.) and SCDHEC Air Pollution Control Regulations 61-62.

Construction and operation permits or exemptions will be required for new nonradiological air emission sources (diesel generators, concrete batch plants etc.) constructed and operated at any SNF facility. The permits will contain operating conditions and effluent limitations for pollutants emitted from the facilities (see Table 7-1).

DOE is currently determining if a NESHAP permit will be required for radiological emissions from any spent nuclear fuel facilities (stacks, process vents, etc.). As described in 40 CFR Part 61.96, if the effective dose equivalent caused by all emissions from facility operations is projected to be less than 1 percent of the 10 millirem per year NESHAP standard, an application for approval to construct under 40 CFR Part 61.07 is not required to be filed. 40 CFR Part 61.96 also allows DOE to use, with prior EPA approval, methods other than EPA standard methods for estimating the source term for use in calculating the projected dose. DOE is currently investigating methods for estimating the transfer, storage and treatment facility source term in accordance with NESHAP requirements to calculate if the emissions would result in an effective dose equivalent of less than the 0.1 millirem per year level. Based on the results of this calculation, DOE will, prior to the start of construction, request EPA approval of the methodology for calculating the projected dose or complete a NESHAP permit application.

Federal Clean Water Act, as amended (33 USC 1251 et seq.); SC Pollution Control Act (SC Code Section 48-1-10 et seq., 1976) (SCDHEC Regulation 61-9.122 et. seq.)

The Federal Water Pollution Act (commonly known as the Clean Water Act), was enacted to "restore and maintain the chemical, physical and biological integrity of the Nation's water." The Clean Water Act prohibits the "discharge of toxic pollutants in toxic amounts" to navigable waters of the United States (Section 101). Section 313 of the Clean Water Act, as amended, requires all branches of the Federal Government engaged in any activity that might result in a discharge or runoff of pollutants to surface waters to comply with Federal, state, interstate, and local requirements.

In addition to setting water quality standards for the Nation's waterways, the Clean Water Act supplies guidelines and limitations (Sections 301-303) for effluent discharges from

point-source discharges and provides authority (Sections 401-402) for the EPA to implement the National Pollutant Discharge Elimination System (NPDES) permitting program pursuant to 40 CFR Part 122 *et seq.*

EPA has delegated primary enforcement authority for the Clean Water Act and the NPDES Permitting Program to SCDHEC for waters in South Carolina. In 1996, SCDHEC, under the authority of the Pollution Control Act (48-1-10 *et seq.*) and Regulation 61-9.122, issued NPDES Permit SC0000175, which addresses wastewater discharges to SRS streams and NPDES permit SCG250162 which address general utility water discharges. The permit contains effluent limitations for physical parameters such as flow and temperature and for chemical pollutants with which the permittee/discharge must comply. DOE will apply for a discharge permit for SNF facilities if the process chosen results in discharges to waters of the State (see Table 7-1).

In Section 402(p) of the Clean Water Act EPA established regulations (40 CFR Part 122.26) for issuing permits for stormwater discharges associated with industrial activity. Accordingly, SCDHEC has issued a General Permit for Storm Water Discharges Associated with Industrial Activities (Permit No. SCR000000) authorizing stormwater discharges to the waters of the State of South Carolina in accordance with effluent limitations, monitoring requirements, and conditions as set forth in the permit. This permit requires preparation and submittal of a Pollution Prevention Plan for all new and existing point source discharges associated with industrial activity. Accordingly, DOE-SR has developed a Storm Water Pollution Prevention Plan (SWPPP) for storm water discharges at SRS. The SRS SWPPP would need to be revised to include pollution prevention measures to be implemented for operation of SNF facilities (See Table 7-1) if industrial activities are exposed to stormwater. SCDHEC has issued a General Permit for stormwater discharges from construction activities that are "Associated with Industrial Activity" (Permit No. SCR100000). An approved plan would be needed that includes

erosion control and pollution prevention measures to be implemented for construction activities.

Section 404 of the Clean Water Act requires that a 404 Permit be issued for discharge of dredge or fill material into the waters of the United States. The authority to implement these requirements has been given to the U.S. Army Corps of Engineers. Section 401 of the Clean Water Act requires certification that discharges from construction or operation of facilities, including discharges of dredged and fill material into navigable waters will comply with applicable water standards. This certification, which is granted by SCDHEC, is a prerequisite for the 404 permit. DOE does not believe that a 404 permit will be required for construction of the SNF facilities.

Federal Safe Drinking Water Act, as amended [42 USC 300 (F) et seq., 40 CFR Parts 100-149]; South Carolina Safe Drinking Water Act (Title 44-55-10 et seq.), State Primary Drinking Water Regulations, (SCDHEC R.61-58)

The primary objective of the Safe Drinking Water Act (42 USC 300), as amended, is to protect the quality of the public water supplies and all sources of drinking water. The implementing regulations, administered by the EPA unless delegated to the States, establish standards applicable to public water systems. They promulgate maximum contaminant levels (including those for radioactivity), in public water systems, which are defined as water systems that serve at least 15 service connections used by year-round residents or regularly serve at least 25 year-round residents. Safe Drinking Water Act requirements have been promulgated by the EPA in 40 CFR Parts 100 through 149. Other programs established by the Safe Drinking Water Act include the Sole Source Aquifer Program, the Wellhead Protection Program, and the Underground Injection Control Program.

EPA has delegated primary enforcement authority to SCDHEC for public water systems in South Carolina. Under the authority of the South Carolina Safe Drinking Water Act (44-55-

10 et seq.), SCDHEC has established a drinking water regulatory program (R.61-58). For radioactive material, the regulations specify that the average annual concentration of manmade radionuclides in drinking water as delivered to the user by such a system shall not produce a dose equivalent to the total body or an internal organ greater than four millirem per year beta-gamma activity. Construction and operation permits will be required for any major new components associated with the SNF facilities. See Table 7-1.

Resource Conservation and Recovery Act, as amended (Solid Waste Disposal Act) (42 USC 6901 et seq.); South Carolina Hazardous Waste Management Act, Section 44-56-30, South Carolina Hazardous Waste Management Regulations (R.61-79.124 et seq.)

The treatment, storage, or disposal of hazardous and nonhazardous waste is regulated under the Solid Waste Disposal Act as amended by the Resource Conservation and Recovery Act (RCRA) and the Hazardous and Solid Waste Amendments of 1984. Pursuant to Section 3006 of the Act, any state that seeks to administer and enforce a hazardous waste program pursuant to RCRA may apply for Environmental Protection Agency authorization of its program. The EPA regulations implementing RCRA (40 CFR Parts 260 through 280) define hazardous wastes and specify their transportation, handling, treatment, storage, and disposal requirements.

The regulations imposed on a generator or a treatment, storage, or disposal facility vary according to the type and quantity of material or waste generated, treated, stored, or disposed of. The method of treatment, storage, or disposal also affects the extent and complexity of the requirements.

Historically, DOE chemically processed spent nuclear fuel to recover valuable products and fissionable materials, and as such, the spent nuclear fuel was not a solid waste under the Resource Conservation and Recovery Act.

World events have resulted in significant changes in DOE's direction and operations. In particular, in April 1992, DOE announced the phase-out of processing for the recovery of special nuclear materials. With these changes, DOE's focus on most of its spent nuclear fuel has changed from processing and recovery of materials to storage and ultimate disposition. This in turn has created uncertainty regarding the regulatory status of some of DOE's spent nuclear fuel relative to the Resource Conservation and Recovery Act.

DOE has initiated discussion with the Environmental Protection Agency on the potential applicability of the Resource Conservation and Recovery Act to spent nuclear fuel. In 1995, an investigation of the applicability of RCRA regulations to a variety of spent fuels and special nuclear materials that were then stored on the SRS (Huggins 1995) concluded, based largely on process knowledge and engineering judgment, that none of the spent fuel in question contained RCRA listed or characteristic material. The evaluated fuel types and cladding materials included aluminum cladding, uranium metal, thorium dioxide, uranium and thorium dioxide powders and pellets, uranium-plutonium powders and pellets, and beryllium oxide powders and pellets. The specific fuels are not necessarily identical to those evaluated in this EIS; however the calculations were conservative and assumed that similar types of fuel or cladding would generally have the same material specifications regardless of where the fuel was fabricated (Huggins 1995). Uranium silicide fuels were not considered in the 1995 evaluation but, based on the general chemical composition (Knight 1993) of uranium silicide fuel and the method used for Toxicity Characteristic Leaching Procedure (TCLP) calculations (Huggins 1995), it does not appear that uranium silicide fuels would qualify as a RCRA hazardous waste.

***The Federal Facility Compliance Act (FFCA)
(42 USC 6921 (et. seq.))***

The FFCA was enacted on October 6, 1992, amended the Resource Conservation Recovery Act. The FFCA waived sovereign immunity for fines and penalties for violations at Federal facilities associated with the management of mixed waste. However, a provision postpones fines and penalties after 3 years for mixed waste storage prohibition violations at DOE sites and requires DOE to prepare plans for developing the required treatment capacity for mixed waste stored or generated at each facility. Each plan must be approved by the host State or the EPA, after consultation with other affected States, and a consent order must be issued by the regulator requiring compliance with the plan. The Federal Facility Compliance Act further provides that DOE will not be subject to fines and penalties for land disposal restriction storage prohibition violations for mixed waste as long as it is in compliance with such an approved plan and consent order and meets all other applicable regulations. This would apply to mixed waste generated as a result of operation of SNF management facilities which are subject to requirements of the Resource Conservation and Recovery Act. On September 20, 1995, the SCDHEC approved, with modification, the Site Treatment Plan for SRS. SCDHEC issued a consent order, signed by DOE, requiring compliance with the plan on September 29, 1995. DOE would be required to notify SCDHEC of new mixed waste streams generated as result of SNF management operations.

***Federal Aviation Act of 1958 (49 USC 1504)
Federal Aviation Administration Regulations
(14 CFR Part 77)***

The Federal Aviation Administration requires that a permit be issued for any structure greater than 200 feet in height which would affect navigable airspace (see Table 7-1). A permit would be required for structures at the SNF site greater than 200 feet in height.

7.1.2 PROTECTION OF BIOLOGICAL, HISTORIC, AND ARCHAEOLOGICAL RESOURCES

The following statutes pertain to protection of endangered and threatened animal and plants.

Endangered Species Act, as amended (16 USC 1531 et seq.)

The Endangered Species Act, as amended, is intended to prevent the further decline of endangered and threatened species and to restore these species and their habitats. The Act is jointly administered by the United States Departments of Commerce and Interior. Section 7 of the Act requires consultation with the Fish and Wildlife Service (Interior) and the National Marine Fisheries Service (Commerce) to determine if endangered and threatened species or their critical habitats are in the vicinity of the proposed action. DOE will comply with the Section 7 Process.

All sites considered for construction of new SNF management facilities are within fenced, disturbed industrial areas. The potential for conditions suitable to support threatened or endangered species does not exist.

Migratory Bird Treaty Act, as amended (16 USC 703 et seq.)

The Migratory Bird Treaty Act, as amended, is intended to protect birds that have common migration patterns between the United States and Canada, Mexico, Japan, and Russia. It regulates the harvest of migratory birds by specifying things such as the mode of harvest, hunting seasons, and bag limits. The Act stipulates that it is unlawful at any time, by any means, or in any manner to "kill...any migratory bird." DOE would be required to consult with the Fish and Wildlife Service regarding impacts to migratory birds and to evaluate ways to avoid or minimize these effects in accordance with the Fish and Wildlife Service Mitigation Policy during construction and operation of SNF management facilities.

Bald and Golden Eagle Protection Act, as amended (16 USC 668-668d)

The Bald and Golden Eagle Protection Act makes it unlawful to take, pursue, molest, or disturb bald and golden eagles, their nests, or their eggs anywhere in the United States (Sections 668, 668c). A permit must be obtained from the U.S. Department of the Interior to relocate a nest that interferes with resource development or recovery operations. All sites considered for the SNF management facilities are within fenced industrial areas without habitat suitable for nesting eagles.

National Historic Preservation Act, as amended (16 USC 470 et seq.)

The National Historic Preservation Act, as amended, provides that sites with significant national historic value be placed on the *National Register of Historic Places*. No permits or certifications are required under the Act. However, if a particular Federal activity could impact an historic property resource, consultation with the Advisory Council on Historic Preservation will usually generate a Memorandum of Agreement, including stipulations that must be followed to minimize adverse impacts. Coordination with the South Carolina State Historic Preservation Officer (SC SHPO) ensures the proper identification of potentially significant sites and the implementation of appropriate mitigative actions. All sites considered for SNF management facilities are within previously disturbed industrial sites.

Archaeological Resource Protection Act, as amended (16 USC 470 et seq.)

This Act requires a permit for any excavation or removal of archaeological resources from public or Native American lands. Excavations must be undertaken for the purpose of furthering archaeological knowledge in the public interest, and resources removed are to remain the property of the United States. Consent must be obtained from the Indian Tribe owning lands on which a resource is located before a permit is is-

sued, and the permit must contain terms or conditions requested by the Tribe.

Native American Grave Protection and Repatriation Act of 1990 (25 USC 3001)

This law directs the Secretary of Interior to assume responsibilities for repatriation of Federal archaeological collections and collections held by museums receiving Federal funding that are culturally affiliated with Native American Tribes. Major actions to be taken under this law include (1) establishing a review committee with monitoring and policy-making responsibilities, (2) developing regulations for repatriation, including procedures for identifying lineal descent or cultural affiliation needed for claims, (3) overseeing museum programs designed to meet the inventory requirements and deadlines of this law, and (4) developing procedures to handle unexpected discoveries of graves or grave goods during activities on Federal or tribal land.

American Indian Religious Freedom Act of 1978 (42 USC 1996)

This Act reaffirms Native American religious freedom under the First Amendment, and sets U.S. policy to protect and preserve the inherent and constitutional right of Native Americans to believe, express, and exercise their traditional religions. The Act requires that Federal actions avoid interfering with access to sacred locations and traditional resources that are integral to the practice of religion.

In conjunction with 1991 studies related to the New Production Reactor, DOE solicited the concerns of Native Americans about religious rights in the Central Savannah River Valley. During this study, three Native American groups -- the Yuchi Tribal Organization, the National Council of Muskogee Creek, and the Indian People's Muskogee Tribal Town Confederacy -- expressed general concerns about SRS and the Central Savannah River Area, but did not identify specific sites as possessing religious significance. The Yuchi Tribal Organization and the National Council of Muskogee

Creek are interested in plant species traditionally used in tribal ceremonies, such as redroot, button snakeroot, and American ginseng (DOE 1991). Redroot and button snakeroot are known to occur on the SRS (Batson, Angerman, and Jones 1985).

In addition, the Savannah River Archaeological Research Program (SRARP) conducted an archeological survey of the preferred APT site in March 1997. The archeological review included potential sites associated with Native American activities or habitat. The resulting SRARP report stated that no archaeological sites present on the preferred site were eligible for nomination to the National Registry of Historical Places and further indicated that SRARP would request from the SC SHPO a determination of no effect from the construction of APT at the preferred site.

7.2 Statutes and Regulations Related to Emergency Planning, Worker Safety, and Protection of Public Health and the Environment

7.2.1 ENVIRONMENTAL PROTECTION

National Environmental Policy Act (NEPA) of 1969, as amended (42 USC 4321 et seq.)

NEPA establishes a national policy promoting awareness of the environmental consequences of human activity on the environment and consideration of environmental impacts during the planning and decisionmaking stages of a project. This Act requires Federal agencies to prepare a detailed statement on the environmental effects of proposed major Federal actions that might significantly affect the quality of the human environment.

This EIS has been prepared in response to NEPA requirements and policies, and in accordance with Council on Environmental Quality (40 CFR Parts 1500 through 1508) and DOE (10 CFR Part 1021) regulations for implementing the procedural provisions of NEPA. It discusses

reasonable alternatives and their potential environmental consequences.

Pollution Prevention Act of 1990 (42 USC 13101 et seq.)

The Pollution Prevention Act of 1990 establishes a national policy for waste management and pollution control that focuses first on source reduction, followed sequentially by environmentally safe recycling, treatment, and disposal. Disposal or releases to the environment should occur only as a last resort. In response, DOE has committed to participation in the Superfund Amendments and Reauthorization Act Section 313, U.S. EPA 33/50 Pollution Prevention Program. The goal for facilities already involved in Section 313 compliance is to achieve by 1997 a 33-percent reduction in the release of 17 priority chemicals from a 1993 baseline. On August 3, 1993, President Clinton issued Executive Order 12856, expanding the 33/50 program such that DOE had to reduce its total releases of all toxic chemicals by 50 percent by December 31, 1999. In addition, DOE is requiring each of its sites to establish site-specific goals to reduce the generation of all waste types.

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Comprehensive Guideline for Procurement of Products Containing Recovered Materials (40 CFR Part 247)

This regulation is issued under the authority of Section 6002 of the Resource Conservation and Recovery Act and Executive Order 12783, which set forth requirements for Federal agencies to procure products containing recovered materials for use in their operations using guidelines established by the EPA. The purpose of these regulations is to promote recycling by using government purchasing to expand markets for recovered materials. RCRA Section 6002 requires that any purchasing agency, when using appropriated funds to procure an item, shall purchase it with the highest percentage of recovered materials practicable. The procurement of materials to be utilized in the construction and operation of SNF management facilities should be conducted in accordance with these regulations.

***Toxic Substances Control Act, as amended
(USC 2601 et seq.) (40 CFR Part 700 et seq.)***

The Toxic Substances Control Act regulates the manufacture, use, treatment, storage, and disposal of certain toxic substances not regulated by the Resource Conservation and Recovery Act or other statutes, particularly polychlorinated biphenyls (40 CFR Part 761), chlorofluorocarbons (40 CFR Part 762), and asbestos (40 CFR Part 763). It is expected that the use of these materials at APT would be limited, or not occur; however, programs and procedures would need to be implemented to address appropriate management and disposal of waste generated as a result of their use.

7.2.2 EMERGENCY PLANNING AND RESPONSE

This section discusses the regulations which address protection of public health, worker safety, and require the establishment of emergency plans and the coordination with local and Federal agencies related to facility operations. DOE Orders generally set forth the programs and procedures required to implement the requirements of these regulations. See Section 7.4.

***Atomic Energy Act of 1954, as amended
(42 USC 2011 et seq.)***

The Atomic Energy Act of 1954 authorizes DOE to establish standards to protect health or minimize dangers to life or property with respect to activities under its jurisdiction. Through a series of Orders, DOE has established an extensive system of standards and requirements to ensure the safe operation of its facilities.

Atomic Energy Act of 1954, as amended (42 USC 2011 et seq.) Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release (10 CFR Part 30.72 Schedule C)

This list is the basis for both the public and private sector to determine if the radiological materials they deal with must have an emergency

response plan for unscheduled releases. It is one of the threshold criteria documents for DOE Emergency Preparedness Hazards Assessments required by DOE Order 151.1, "Comprehensive Emergency Management System." An emergency response plan addressing SNF operations would need to be promulgated in accordance with this regulation.

Reorganization Plan No. 3 of 1978, Public Health and Welfare (42 USC 5121 et seq.), Emergency Management and Assistance (44 CFR Part 1-399)

These regulations generally include the policies, procedures and set forth the responsibilities of the Federal Emergency Management Agency, the Nuclear Regulatory Commission, and the Department of Energy for implementing a Federal Emergency Preparedness Program including radiological planning and preparedness. An emergency response plan, including radiological planning and preparedness for SNF management operations, would need to be prepared and implemented, in accordance with this regulation.

***Emergency Planning and Community Right-to-Know Act of 1986 (42 USC 11001 et seq.)
(also known as "SARA Title III")***

The Emergency Planning and Community Right-to-Know Act of 1986 requires emergency planning and notice to communities and government agencies of the presence and release of specific chemicals. EPA implements this Act under regulations found at 40 CFR Parts 355, 370, and 372. Under Subtitle A of this Act, Federal facilities provide various information (such as inventories of specific chemicals used or stored and releases that occur from these facilities) to the State Emergency Response Commission and the Local Emergency Planning Committee to ensure that emergency plans are sufficient to respond to unplanned releases of hazardous substances. Implementation of the provisions of this Act began voluntarily in 1987, and inventory and annual emissions reporting began in 1988. In addition, DOE requires compliance with Title III as a matter of Departmental policy. The requirements for this Act were

promulgated by EPA in 40 CFR Parts 350 through 372. The SRS submits hazardous chemical inventory reports to the SCDHEC. The chemical inventory could change depending on the alternative(s) DOE implemented; however, subsequent reports would reflect any change to the inventory.

Transportation of Hazardous Materials (49 USC 5101 et seq.); Hazardous Materials Tables & Communications, Emergency Response Information Requirements (49 CFR Part 172)

The regulatory requirements for marking, labeling, placarding, and documenting hazardous materials shipments are defined in this regulation. It also specifies the requirements for providing hazardous material information and training. Materials shipped to and from SNF management facilities would be required to comply with these regulations.

Comprehensive Environmental Response, Compensation, and Liability Act of 1980, as amended (42 USC 9601 et seq.) National Oil and Hazardous Substance Contingency Plan (40 CFR Part 300 et seq.)

More popularly known as "Superfund," the Act and implementing regulations provide the needed general authority for Federal and state governments to respond directly to hazardous substances incidents. The regulations require reporting of spills, including radioactive, to the National Response Center. SNF management operations would be required to comply with these regulations in the event of spills of hazardous materials at SNF facilities. DOE Orders generally set forth the programs for development of internal procedures for implementing the regulations.

Occupational Safety and Health Act of 1970, as amended (29 USC 651 et seq.); Occupational Safety and Health Administration Emergency Response, Hazardous Waste Operations and Worker Right to Know (29 CFR Part 1910 et seq.)

The Occupational Safety and Health Act (29 USC 651) establishes standards to enhance safe

and healthful working conditions in places of employment throughout the United States. The Act is administered and enforced by the Occupational Safety and Health Administration, a U.S. Department of Labor agency. While the Occupational Safety and Health Administration and EPA both have a mandate to reduce exposures to toxic substances, the Occupational Safety and Health Administration's jurisdiction is limited to safety and health conditions that exist in the workplace environment. In general, under the Act, it is the duty of each employer to furnish all employees a place of employment free of recognized hazards likely to cause death or serious physical harm. Employees have a duty to comply with the occupational safety and health standards and all rules, regulations, and orders issued under the Act. The Occupational Safety and Health Administration regulations (29 CFR) establish specific standards telling employers what must be done to achieve a safe and healthful working environment. This regulation sets down the Occupational Safety and Health Administration requirements for employee safety in a variety of working environments. It addresses employee emergency and fire prevention plans (Section 1910.38), hazardous waste operations and emergency response (Section 1910.120), and hazards communication (Section 1910.1200) that enables employees to be aware of the dangers they face from hazardous materials at their workplace. DOE places emphasis on compliance with these regulations at its facilities and prescribes through DOE Orders the Occupational Safety and Health Act standards that contractors shall meet, as applicable to their work at Government-owned, contractor-operated facilities. DOE keeps and makes available the various records of minor illnesses, injuries, and work-related deaths required by Occupational Safety and Health Administration regulations.

Noise Control Act of 1972, as amended (42 USC 4901 et seq.)

Section 4 of the Noise Control Act of 1972, as amended, directs all Federal agencies to carry out "to the fullest extent within their authority" programs within their jurisdictions in a manner

that furthers a national policy of promoting an environment free from noise that jeopardizes health and welfare.

7.3 Executive Orders

The following executive orders would be in effect for the construction and operation of the APT. DOE Orders generally set forth the programs and procedures required to implement the requirements of the orders.

Executive Order 11514 (Protection and Enhancement of Environmental Quality)

Executive Order 11514 requires Federal agencies to monitor and control their activities continually to protect and enhance the quality of the environment and to develop procedures to ensure the fullest practicable provision of timely public information and understanding of Federal plans and programs with environmental impact to obtain the views of interested parties.

Executive Order 11988 (Floodplain Management)

Executive Order 11988 requires Federal agencies to establish procedures to ensure that the potential effects of flood hazards and floodplain management are considered for any action undertaken in a floodplain and that floodplain impacts be avoided to the extent practicable.

Executive Order 11990 (Protection of Wetlands)

Executive Order 11990 requires Government agencies to avoid any short- and long-term adverse impacts on wetlands wherever there is a practicable alternative.

Executive Order 12856 (Right-to-Know Laws and Pollution Prevention Requirements)

Executive Order 12856 requires all Federal agencies to reduce the toxic chemicals entering any waste stream. This order also requires Federal agencies to report toxic chemicals entering waste streams; improve emergency planning, re-

sponse, and accident notification; and encourage clean technologies and testing of innovative prevention technologies.

Executive Order 12898 (Environmental Justice)

Executive Order 12898 requires Federal agencies to identify and address disproportionately high and adverse human health or environmental effects of its programs, policies, and activities on minority and low-income populations.

Executive Order 12902 (Energy Efficiency and Water Conservation at Federal Facilities)

Executive Order 12902 requires Federal agencies to develop and implement a program for conservation of energy and water resources.

7.4 DOE Regulations and Orders

Through the authority of the Atomic Energy Act, DOE is responsible for establishing a comprehensive health, safety, and environmental program for its facilities. The regulatory mechanisms through which DOE manages its facilities are the promulgation of regulations and the issuance of DOE Orders. Table 7-2 lists the major DOE Orders applicable to the construction and operation of SNF management facilities.

The DOE regulations address such areas as energy conservation, administrative requirements and procedures, nuclear safety, and classified information. For the purposes of this EIS, relevant regulations include 10 CFR Part 820, *Procedural Rules for DOE Nuclear Facilities*; 10 CFR Part 830, *Nuclear Safety Management; Contractor and Subcontractor Activities*; 10 CFR Part 835, *Occupational Radiation Protection*; 10 CFR Part 1021, *Compliance with NEPA*; and 10 CFR Part 1022, *Compliance with Floodplains/Wetlands Environmental Review Requirements*. DOE has enacted occupational radiation protection standards to protect DOE and its contractor employees. These standards are set forth in 10 CFR Part 835, *Occupational Radiation Protection*; the rules in this part es-

establish radiation protection standards, limits, and program requirements for protecting individuals from ionizing radiation resulting from the conduct of DOE activities, including those conducted by DOE contractors. The activity may be, but is not limited to, design, construc-

tion, or operation of DOE facilities. These regulations would be in effect for the construction and operation of any facilities associated with the production and management of tritium. DOE Orders generally set forth policy and the programs and internal procedures for implementing those policies.

Table 7-2. DOE Orders and Notices relevant to spent nuclear fuel management.

DOE Order	DOE Orders
151.1	Comprehensive Emergency Management System
225.1	Accident Investigations
231.1	Environment, Safety, and Health Reporting
232.1	Occurrence Reporting and Processing of Operations Information
420.1	Facility Safety
425.1	Startup and Restart of Nuclear Facilities
430.1	Life-Cycle Asset Management
440.1	Worker Protection Management for DOE Federal and Contractor Employees
441.1	DOE Radiological Health and Safety Policy
441.2	Extension of DOE 441.1 (9-19-96)
441.3	Extension of DOE 441.1 (9-17-97)
451.1A	National Environmental Policy Act Compliance Program
460.1A	Packaging and Transportation Safety
460.2	Departmental Materials and Packaging Management
470.1	Safeguards and Security Program
471.1	Identification and Protection of Unclassified Controlled Nuclear Information
471.2A	Information Security Program
472.1B	Personnel Security Activities
1270.2B	Safeguards Agreement with the International Atomic Energy Agency
1300.2A	Department of Energy Technical Standards Program
1360.2B	Unclassified Computer Security Program
3790.1B	Federal Employee Occupational Safety and Health Program
4330.4B	Maintenance Management Program
4700.1	Project Management System
5400.1	General Environmental Protection Program
5400.3	Hazardous and Radioactive Mixed Waste Program
5400.5	Radiation Protection of the Public and the Environment
5480.4	Environmental Protection, Safety, and Health Protection Standards
5480.17	Site Safety Representatives
5480.19	Conduct of Operations Requirements for DOE Facilities
5480.20A	Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities
5480.21	Unreviewed Safety Questions
5480.22	Technical Safety Requirements
5480.23	Nuclear Safety Analysis Reports
5480.25	Safety of Accelerator Facilities
5480.27	Equipment Qualification for Reactor and Nonreactor Nuclear Facilities
5484.1	Environmental Protection, Safety, and Health Protection Information Reporting Requirements
5630.12A	Safeguards and Security Inspection and Evaluation Program
5632.1C	Protection and Control of Safeguards and Security Interests
5633.3B	Control and Accountability of Nuclear Materials
5660.1B	Management of Nuclear Materials
5700.6C	Quality Assurance
5820.2A	Radioactive Waste Management
6430.1A	General Design Criteria

Table 7-2. (Continued)

DOE Standards	
1020-94	Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities
1021-94	Natural Phenomena Hazards Performance Categorization Criteria for Structure, Systems, and Components
1024-92	Guidelines for Use of Probabilistic Seismic Hazard Curves at Department of Energy Sites
1027-92	Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23 Nuclear Safety Analysis Reports
3009-94	Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports
3011-94	DOE Standard Guidance for Preparation of DOE 5480.22 and DOE 5480.23 Implementation Plans

References

- Batson, W. T., J. S. Angerman, and J. T. Jones, 1985, *Flora of the Savannah River Plant: An Inventory of the Vascular Plants on the Savannah River Plant, South Carolina*, Savannah River Plant National Environmental Research Park Program, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1991, American Indian Religious Freedom Act Compliance at the Savannah River Site, Savannah River Operations Office, Aiken, South Carolina.
- Huggins, L. B., 1995, *RCRA Characterization of Nuclear Materials in Inventory (NMII) Within the Excess Facilities and Reactor Fuel Storage Program Division*, Westinghouse Savannah River Company, Aiken, South Carolina.
- Knight, R. W., 1993, *Observations in the Manufacture of Aluminum-Based Research Reactor Fuel Elements*, Martin Marietta, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

APPENDIX A

TECHNOLOGY DESCRIPTIONS

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APPENDIX A. TECHNOLOGY DESCRIPTIONS

This appendix describes the technology options that the U.S. Department of Energy (DOE) considered for implementing the spent nuclear fuel (SNF) management alternatives. As described in Chapter 2, DOE consolidated many of these options and eliminated two from further consideration. This appendix addresses each technology option. The options are grouped according to the technology to which they would apply.

A.1 New Packaging Technologies

A.1.1 DIRECT DISPOSAL

As the first step in the Direct Disposal process, the shipping cask would be received and unloaded in the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility. The spent nuclear fuel would be placed in lag storage and the cask would be decontaminated and returned for reuse. The fuel would remain in lag storage until it was ready to be conditioned and repackaged for road-ready storage.

Conditioning activities prior to repackaging would include decanning canned fuel, cropping fuel assemblies to eliminate most of the nonfuel structural components, and limited characterization. The characterization would involve reviewing records, weighing and visually inspecting the fuel, and conducting gamma spectrometry and tests for cladding integrity. In some cases, more complete characterization could be necessary and could result in samples being taken for additional analysis. Failed fuel and other special case fuel would be recanned.

The SNF and failed fuel in cans then would be placed in canisters for road-ready storage. The fuel could be loaded into the canisters in a variety of positions, depending on the size of the fuel and its uranium content. For the Direct Disposal option, DOE could use a 24-inch (61-centimeter) diameter canister available in 5-, 10-, or 16-foot (1.5-, 3-, or 4.9-meter) lengths. Metal frames (called baskets) would be inserted

into the canister to hold the fuel in a fixed position. For materials test reactor-like fuels (Group B), which make up about 97 percent of the volume of the aluminum SNF inventory, each basket would hold 16 fuel assemblies; a 10-foot (3-meter) canister could hold four baskets vertically stacked. Therefore, one 10-foot (3-meter) long canister would hold up to 64 materials test reactor-like fuel assemblies. The amount of fissile mass in the fuel could limit the length of the canister and decrease the number of fuel assemblies loaded into each canister. Dry storage space for about 1,100 24-inch (61-centimeter) diameter canisters would be needed for the Direct Disposal technology.

Before sealing the canisters, the assemblies would be vacuum-dried to remove free water. Water could contribute to continued corrosion of the fuel and to the buildup of hydrogen gas which can be generated by radiolytic decomposition of the water and by metal corrosion. Group A fuels, which are uranium or thorium metal, are more reactive than other fuels and would need more extensive drying to remove the bound water. Hot vacuum drying has been effective in eliminating bound water. However, including hot vacuum drying capability in the Transfer and Storage Facility for the small amount of Group A material (approximately 4 cubic yards [3 cubic meters]) would require a large expenditure that could be disproportionate to the benefit.

Depending on the design of the Transfer and Storage Facility, the canisters could be placed in storage singly, in storage overpacks singly or in groups, or in shipping casks (for storage) singly or in groups. Regardless of design, the fuel would be considered road-ready because no further characterization, conditioning, or other handling would be necessary before shipment. The canisters could require packaging into shipping casks, and they could need venting (to relieve buildup of hydrogen) before shipping.

Approximately 70 percent of the volume of the aluminum-based SNF to be managed at SRS would be highly enriched uranium, which would present special criticality considerations. In addition, most of the fuel considered in this EIS is aluminum-based and thus subject to more rapid corrosion (and loss of the spacing that keeps the uranium in the fuel from undergoing inadvertent criticality) than the more robust commercial or naval fuels. Finally, research reactor fuel generally has experienced lower burnup than commercial fuel, providing greater potential nuclear reactivity. Therefore, DOE proposes to address criticality by (1) a conservative limitation on the amount of fissile material in the waste package, (2) use of neutron absorbers in the fuel baskets to poison the fission chain reaction, and (3) basket and canister design to maintain subcritical geometries. For planning purposes, DOE currently limits the fissile material content to 31.8 pounds (14.4 kilograms) of highly enriched uranium per canister (DOE 1996). This limitation is based on conservative assumptions to meet current Nuclear Regulatory Commission (NRC) requirements for geologic disposal (10 CFR 60.131). DOE believes there is a good technical basis for increasing the fissile material allowance for the canisters and might do so if regulations change. Figure A-1 shows the Direct Disposal process flow.

A.1.2 DIRECT CO-DISPOSAL

From the SRS perspective, Direct Co-Disposal is identical to Direct Disposal, except 17-inch (43-centimeter) by 10-foot (3-meter) canisters would be used. The canisters would be shipped to the repository in shipping casks and repackaged into repository packages in the space among five 24-inch (61-centimeter) by 10 feet (3-meter) high-level waste canisters (see text box on page 2-5). The benefit of Direct Co-Disposal over Direct Disposal would be that little additional repository space would be needed. Because of the smaller diameter canister, approximately 1,400 dry storage spaces would be needed at the SRS for the Direct Co-Disposal technology.

A.2 New Processing Technologies

A.2.1 MELT AND DILUTE

With the Melt and Dilute technology, the shipping cask would be received and unloaded and SNF would be characterized and stored as described for Direct Disposal.

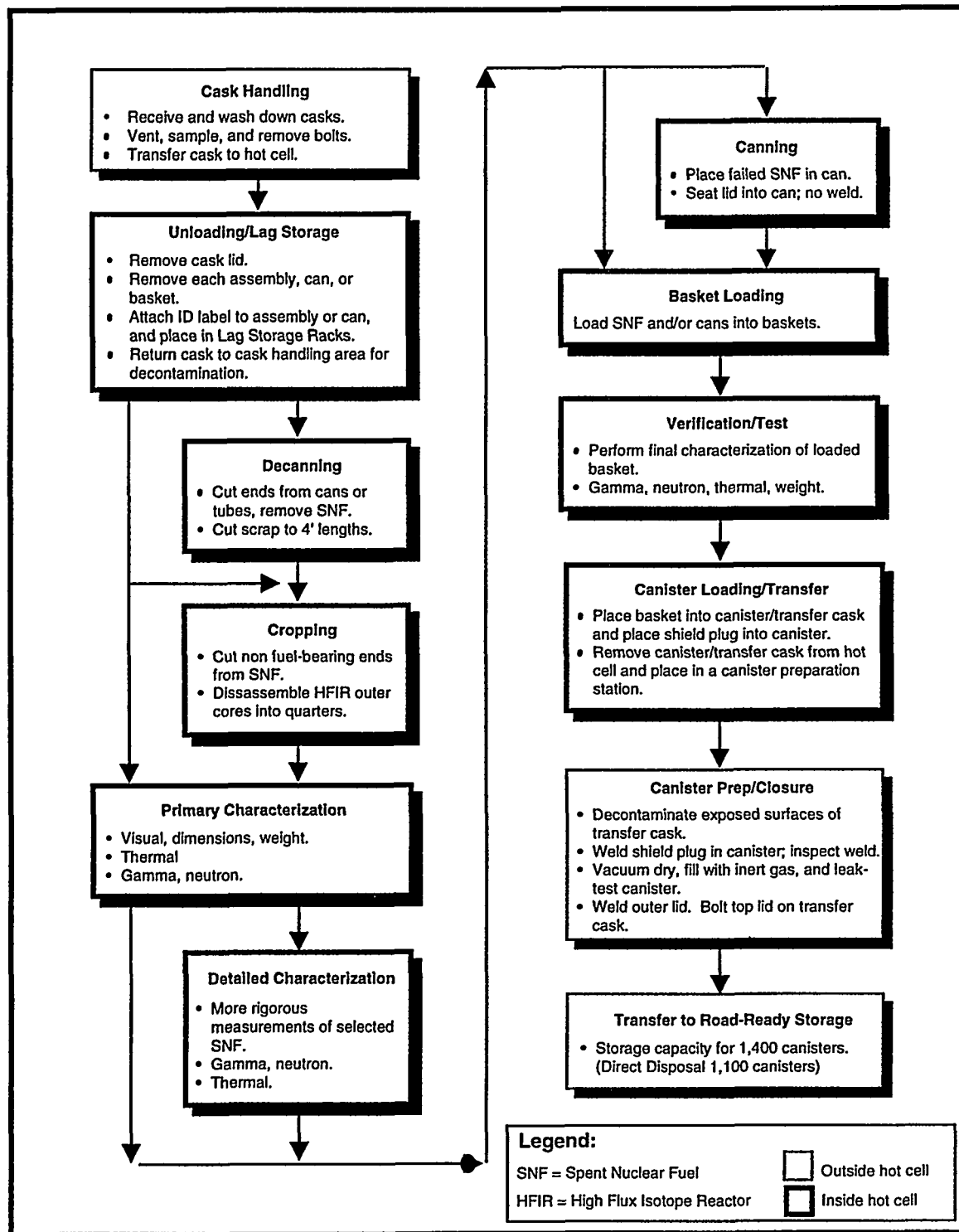
The fuel assemblies would be placed in an induction-heated melter with additions of depleted uranium and aluminum as needed to meet fissile enrichment and alloy composition specifications (see text box-Control of Melt and Dilute Process and Product Characteristics on page A-4). The melt would be contained in a crucible within the melter. The molten metal would be sampled to determine uranium-235 content and alloy composition. Adjustments to the uranium enrichment or alloy composition then could be made.

The adjusted melt would be cast into a form of approximately 16 inches in diameter and 33 inches maximum length. After cooling, the fuel would be loaded into baskets, then loaded into the canisters. The canisters would be evacuated, filled with an inert gas, sealed by welding, and transferred to road-ready storage.

About 400 canisters of the Melt and Dilute product would be produced for dry storage to be loaded as one per co-disposal package for repository disposal.

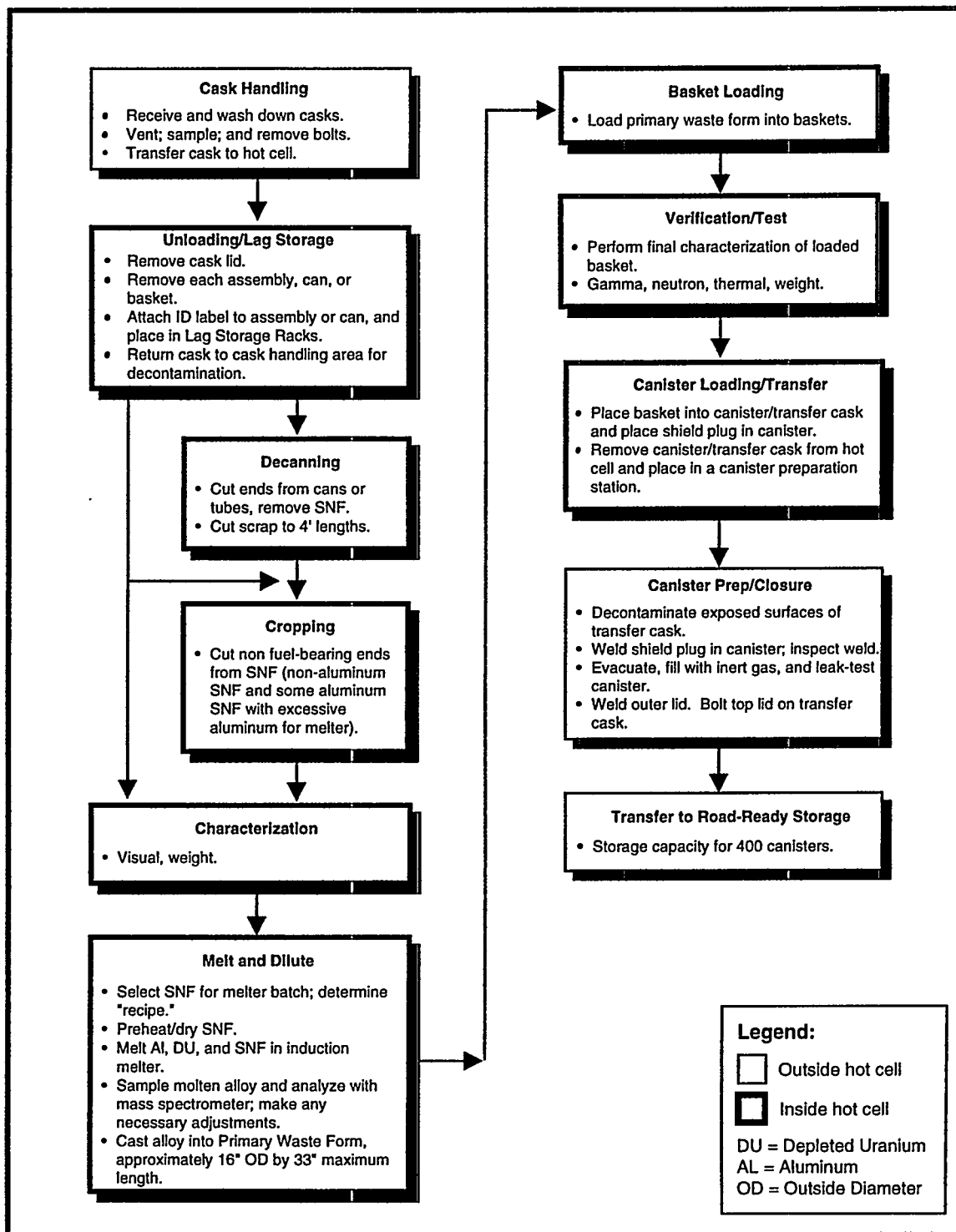
The Melt and Dilute process is a simpler technology than many of the others, especially for metal fuels. An offgas system would capture the volatile and semivolatile fission products. Oxide and silicide fuels would be reduced to metal during the melting process.

Figure A-2 shows the Melt and Dilute process flow diagram.



NW SNF EIS/PubsOnly/SNFApp_a/Grfx_apu/FA-1.ppt

Figure A-1. Direct Disposal/Direct Co-Disposal process flow diagram.



NW SNF EIS/PubsOnly/SNF_App_A/Grfx_APA/FA-2.ppt

Figure A-2. Melt and Dilute process flow diagram.

Control of Melt and Dilute Process and Product Characteristics

The Melt and Dilute treatment allows adjustment of product composition and uranium-235 enrichment to meet process and disposal requirements. Melter temperatures below about 1,000°C, needed to maintain control of crucible interactions and offgas volumes, depend on uranium-aluminum contents of the melt. A candidate alloy composition (13.2 wt percent uranium) melts at 646°C, with melter temperature in the range 750 to 850°C projected for representative operations. Dilution of uranium-235 from original concentrations of as high as 93 percent down to 20 percent by addition of depleted uranium renders the melt product unsuitable for weapons use and reduces its nuclear criticality potential; lower enrichments (typically 2 to 5 percent uranium-235) further reduce criticality to the equivalent of commercial SNF.

Increased uranium content due to the addition of depleted uranium is offset by aluminum additions to maintain low melter temperatures. Dilution to 20 percent uranium-235 requires relatively small depleted uranium and aluminum additions, but dilution to lower enrichment levels requires significantly greater depleted uranium and compensating aluminum additions.

Volume increases of the final melt product due to the depleted uranium and aluminum additions result in larger numbers of waste canisters for disposal. For a product composition of 13.2 wt percent uranium, product volume is affected by final uranium-235 enrichment levels as follows:

Enrichment level, percent U-235	Number of waste canisters ^a
20	400
5	1,234
2	1,796

a. For representative inventory of processed aluminum-SNF assemblies, assuming 0.276 m³ melt product per canister.

At the candidate alloy composition, the melt solidifies to a uniform, relatively stable microstructure of uranium-aluminum phases. Although more reactive in aqueous environments than commercial uranium oxide fuels or high-level waste glass, the melt product is well suited to characterization of reactions with waste package and geologic materials important for long-term projections of waste form behavior in a geologic repository.

A.2.2 PRESS AND DILUTE

With the Press and Dilute technology, the shipping cask would be received and unloaded in the Transfer, Storage and Treatment Facility, and the SNF would undergo a limited characterization involving records review, visual inspection, and gamma spectrometry. In some cases, more complete characterization could be necessary and could result in samples being taken for additional analysis elsewhere. The characterization data would be used to determine the amount of depleted uranium needed to meet dilution requirements (if any).

The fuel assemblies would be cropped to eliminate most of the nonfuel structural components and reduce storage space. The fuel assemblies would be vacuum-dried to remove free water.

The dried assemblies would be placed in a mechanical press for compaction. The pressed spent nuclear fuel would be layered with depleted uranium and pressed again to lock the pieces together. Layering would continue to the limits imposed by the canister dimensions. The shape of the pressed assembly would be determined by future research but would be optimized to reduce free space in the canister. Free space could result in the intrusion of a moderator (e.g., water), thereby changing the assumptions under which nuclear safety calculations were performed. The final shape of the waste form could be cylindrical (from molds) or stacked disks.

Finally, the pressed fuel form would be placed into 17-inch (43-centimeter) diameter canisters, which would be filled with an inert gas and

welded closed. The pressing operation and the canister loading operation would be controlled to limit the fissile material in the canisters, in accordance with nuclear criticality considerations. The number of canisters produced for dry storage would be about 630 to be loaded as one per co-disposal overpack for repository disposal.

The primary advantage of Press and Dilute technology is its simplicity. However, the variable sizes of Group C SNF might make the technology unsuitable for those fuels without special disassembly before compaction. Particulate fuels (Group D) would not be amenable to pressing. Figure A-3 shows the Press and Dilute process flow diagram.

A.2.3 CHOP AND DILUTE

In the Chop and Dilute treatment, the shipping cask would be received and unloaded, and the SNF would undergo a limited characterization as described in Section A.2.2 Press and Dilute. The fuel assemblies would be cropped to eliminate most of the nonfuel structural components and reduce storage space. The assemblies would be vacuum-dried to remove free water.

The dried assemblies would be fed into a shredder. Similarly shredded depleted uranium-aluminum alloy would be combined with the shredded fuel to produce a mix of reduced enrichment. The shredded fuel would be placed into 17-inch (43-centimeter) diameter canisters, which would be filled with inert gas and welded closed. The canister loading would be controlled to limit the amount of fissile material in the canisters in accordance with nuclear criticality requirements. The number of canisters produced for dry storage and repository disposal would be about the same as for the Press and Dilute process (630).

The material resulting from Chop and Dilute would not be homogeneous and would result in a considerable amount of free space in each canister. The free space would contribute to an increase in the number of canisters required and could increase vulnerability to a nuclear criticality. In addition, the material could be pyrophoric. Because of these difficulties with Chop and Dilute, DOE considers it to be the least attractive of the three dilution technologies (Melt and Dilute, Press and Dilute, and Chop and Dilute).

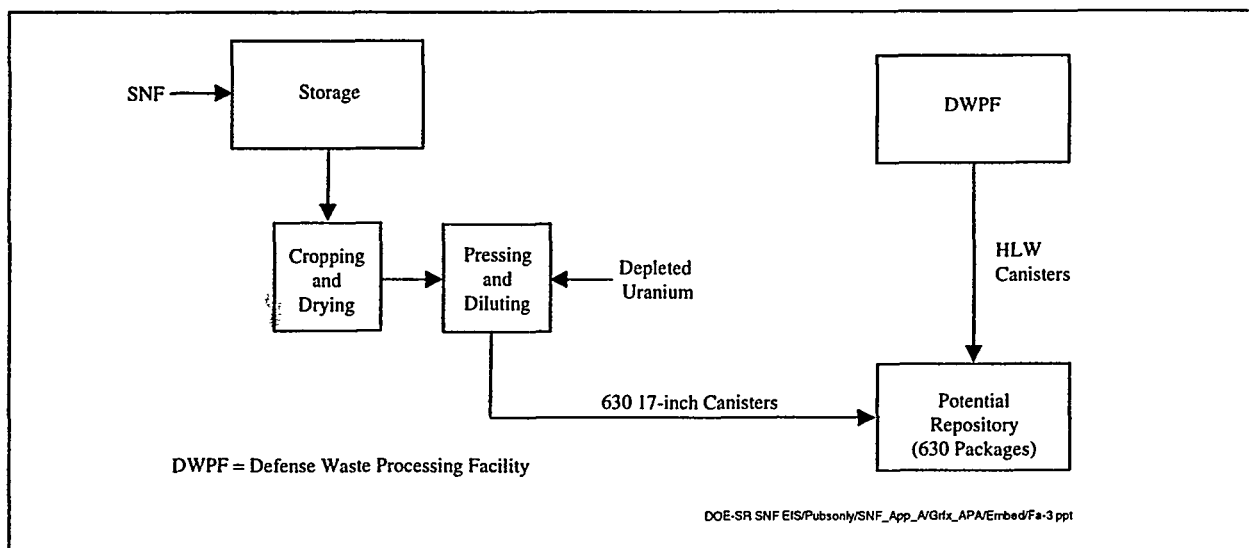


Figure A-3. Press and Dilute process flow diagram.

A.2.4 DISSOLVE AND VITRIFY

The SNF would be cropped and charged to an electrolytical dissolver similar to that used in H Canyon. The electrolyte solution would be nitric acid saturated with boric acid. The process would operate in a batch mode to ensure criticality control. Depleted uranium would be added, as needed, to reduce the uranium-235 enrichment to approximately 5 percent.

The dissolver solution would be transferred to a holding tank for chemical and radiological analyses to determine the need for any adjustments prior to the vitrification step. The solution then would be transferred to an electrically-heated melter, along with glass-forming chemicals. Several dissolver batches could be melted at once. The resulting molten glass, having been preanalyzed in the holding tank, should be of sufficient quality to be poured into canisters similar to those used at the Defense Waste Processing Facility. About 1,350 canisters would be produced for emplacement in about 270 repository waste packages.

Although DOE could perform dissolution using the existing equipment at H Canyon, the analysis in this EIS assumes the construction of a new

Dissolve and Vitrify facility. Figure A-4 shows the Dissolve and Vitrify process flow diagram.

A.2.5 GLASS MATERIAL OXIDATION AND DISSOLUTION SYSTEM

The Glass Material Oxidation and Dissolution System (GMODS) converts SNF directly to borosilicate glass using a batch process. Criticality concerns are addressed by diluting the uranium-235 enrichment with depleted uranium and using boron oxide as a dissolving agent (boron is a neutron poison). Although the addition of depleted uranium and glass frit adds to the mass, the high-density, monolithic glass still would provide a smaller volume for dry storage than would Direct Co-Disposal.

The principal piece of equipment for GMODS would be an induction-heated cold-wall melter, which in commercial use converts corrosive or high-melting metals to ultrapure materials. The melter would be charged with a molten glass consisting of lead oxide and boron oxide. The lead oxide converts the metals in the SNF to oxides; oxides and amorphous materials tend to dissolve in molten glass, but metals do not. Boron oxide is a common agent for dissolving oxides into glass (e.g., welding slag). A problem

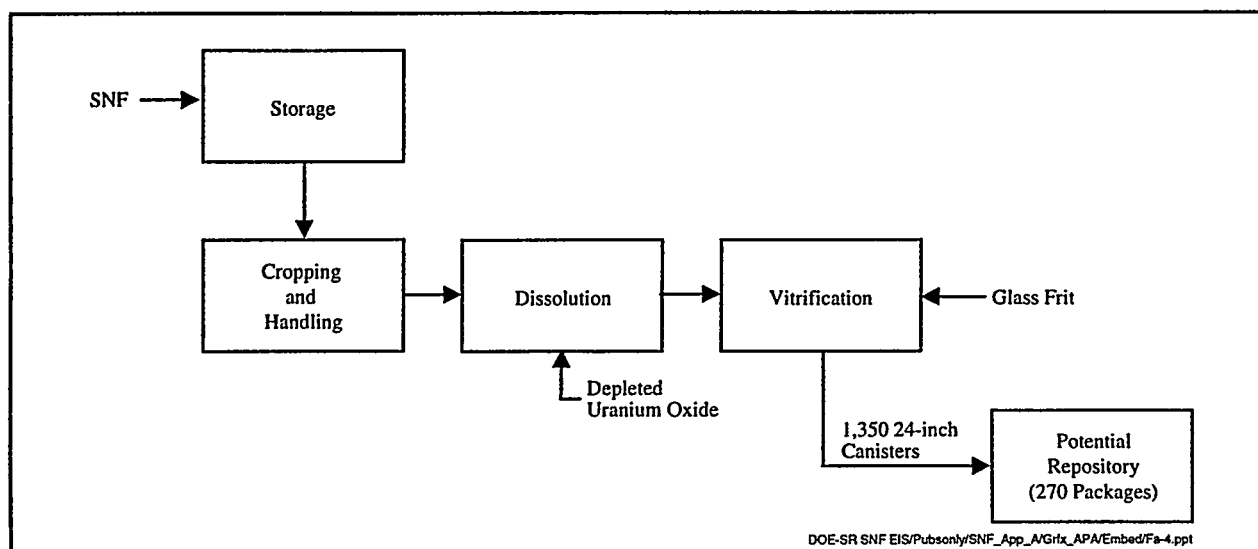


Figure A-4. Dissolve and Vitrify process flow diagram.

with using lead oxide is its corrosivity, which could affect the service life of the melter.

As the SNF is fed into the melter, the aluminum, uranium, and other metals would be oxidized and dissolved in the molten glass. Uranium oxides and other oxides would be directly dissolved. The oxidation of the metals converts the lead oxide to metallic lead, which sinks to the bottom of the melter.

The resulting glass mixture would not have qualities necessary for long-term durability, so silicon oxide (glass frit) additions would be necessary to increase the durability. The silicon oxide would not be part of the initial melter charge because its properties are not conducive to rapid oxidation-dissolution of SNF. Unreduced lead oxide could limit the durability of the glass, and increase volume, so carbon would be added to the melt to reduce the excess lead oxide.

The glass melt would be decanted from the melter and formed into glass marbles. For criticality and other practical reasons, the batch melts using GMODS would not be large enough to fill a 24-inch (61-centimeter) diameter canister. Therefore, the glass marbles would be

stored and remelted, allowing a continuous pour to fill several 24-inch diameter canisters at a time. The GMODS process would produce typically about 1,350 canisters for emplacement in about 270 repository packages.

After decanting the glass, the melter would be recharged with boron oxide and, if necessary, lead oxide. Oxygen would be piped into the system to convert the metallic lead at the bottom of the melter back to lead oxide. Lead would be an oxygen carrier that did not leave the system.

Radioactive offgases produced during this process would be filtered and treated appropriately. Figure A-5 shows the Glass Material Oxidation and Dissolution process flow diagram.

A.2.6 PLASMA ARC TREATMENT

The Plasma Arc Treatment technology uses a plasma torch to melt and oxidize the SNF in conjunction with depleted uranium oxide and other ceramic-forming materials as necessary. The fuel would be fed into the process with minimal sizing or pretreatment. The plasma arc would cut the fuel assemblies into small pieces and heat the fuel to temperatures as high as

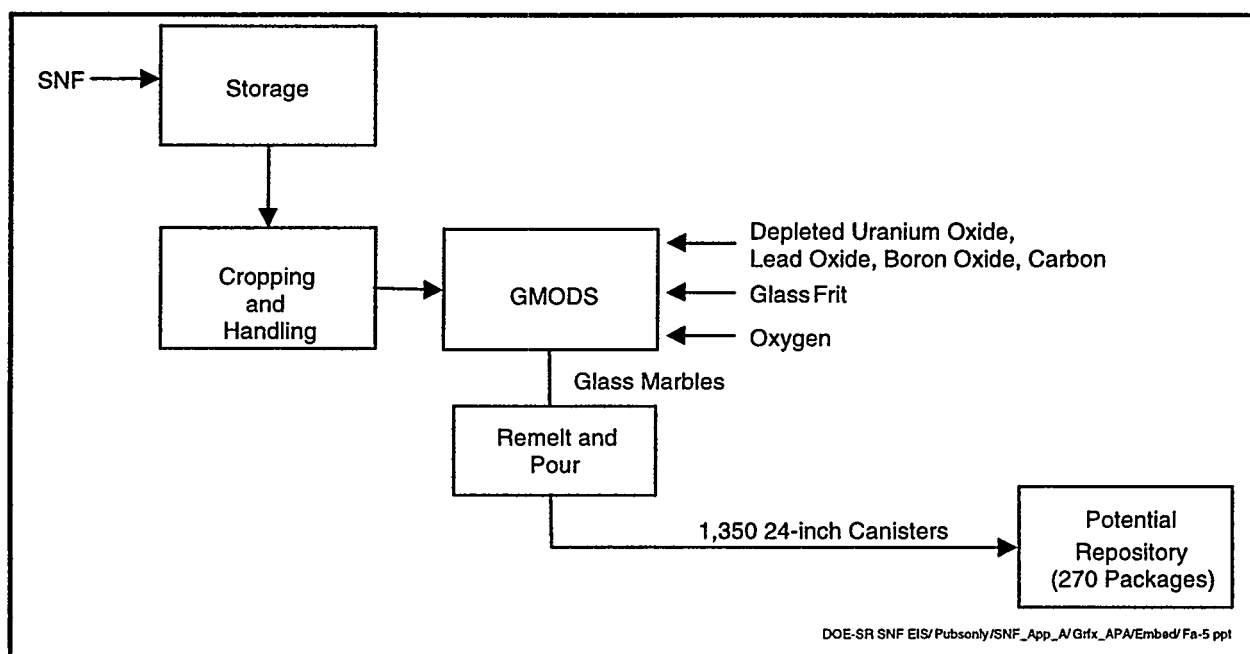


Figure A-5. Glass Material Oxidation and Dissolution process flow diagram.

1,600°C (2,900°F) to melt and oxidize it in a rotating furnace. Ceramic material would be added as necessary with the mixture homogenized by the torch. When melting and oxidation were complete, the rotating furnace would slow and the melt would fall into molds prepared to receive it.

Some types of SNF might not require the addition of ceramic material to the process because the oxidation would produce a robust ceramic form from the fuel itself. Many metallic fuels would, however, need some ceramic addition. Depleted uranium could be added to the process in almost any form to reduce the uranium-235 enrichment. Criticality issues would be addressed by limiting the process to batch runs of preselected quantities of fissile material, by the addition of the depleted uranium, and by the addition of neutron poisons if necessary. The Plasma Arc treatment would produce about 490 canisters to be contained in 98 repository packages.

As with all processes that dissolve or melt the SNF, the Plasma Arc Treatment would produce radioactive offgases. These gases would be filtered and treated by appropriate means, with the filter and treatment media recycled into the

plasma arc furnace for incorporation into the ceramic product. Figure A-6 shows the Plasma Arc Treatment process flow diagram.

A.2.7 ELECTROMETALLURGICAL TREATMENT

The Electrometallurgical Treatment process would adapt a technology under development at the Argonne National Laboratory for processing Experimental Breeder Reactor-II fuel and blanket assemblies. The process has been demonstrated for the stainless steel-clad uranium alloy fuels used in this reactor. The electrorefining process employs a technology used in industry to produce pure metals from impure metal feedstock. The feasibility of the Electrometallurgical Treatment for aluminum-based fuels has been tested in the laboratory and is theoretically possible as conducted in the following two stages. An electrorefiner facility is available at the Idaho National Engineering and Environmental Laboratory to test development concepts.

Preparation

Before electrorefining, the fuel would be cropped and the end fittings discarded. The fuel assemblies would be compacted and melted with

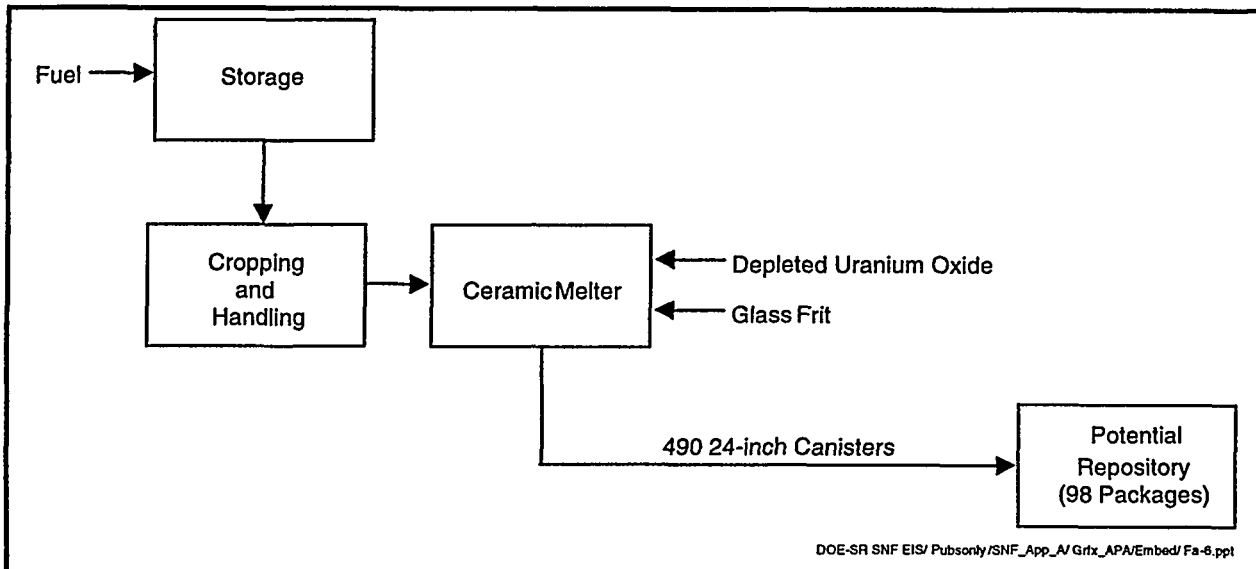


Figure A-6. Plasma Arc Treatment process flow diagram.

silicon added to the melt to complex the uranium and enhance its separation from aluminum in the subsequent electrorefining step.

Melting would vaporize many volatile fission products (e.g., cesium, rubidium, bromine, iodine, xenon, and krypton), which, with the exception of the noble gases, would be captured in a fibrous aluminosilicate trap. The molten fuel would be poured into ingots that would become the anodes for processing in the next step in the electrorefiner.

Aluminum Separation

The electrorefining process would first use a lithium fluoride-potassium fluoride electrolyte to separate the aluminum from the anode. Aluminum and alkaline earth fission products would dissolve out of the ingot; the aluminum would form a soluble compound of potassium aluminum hexafluoride (K_3AlF_6), which would travel to the cathode where it would be reduced to pure aluminum metal. The alkaline earth fission products would remain in the electrolyte. The aluminum deposits on the cathode would be continually scraped off and collected.

Because some electrolyte salts would be entrained with the aluminum, the aluminum would be melted to separate the aluminum from the salts. The melt would cool below the melting point of the aluminum, and the salts would be poured off and recycled. The aluminum would be disposed as low-level waste.

Uranium Separation

After essentially all aluminum was removed from the anode, the actinides (primarily uranium), rare earths, and noble metals would remain. The anode would be placed in a second refiner that used lithium fluoride, potassium fluoride, and uranium trifluoride salts as electrolytes. The uranium in the anode present as a uranium silicide would be oxidized to uranium trifluoride and transported to the cathode where it would be reduced to uranium metal. The ura-

nium metal deposits would be collected and separated from electrolyte salts as was the aluminum.

Salt Scrubbing

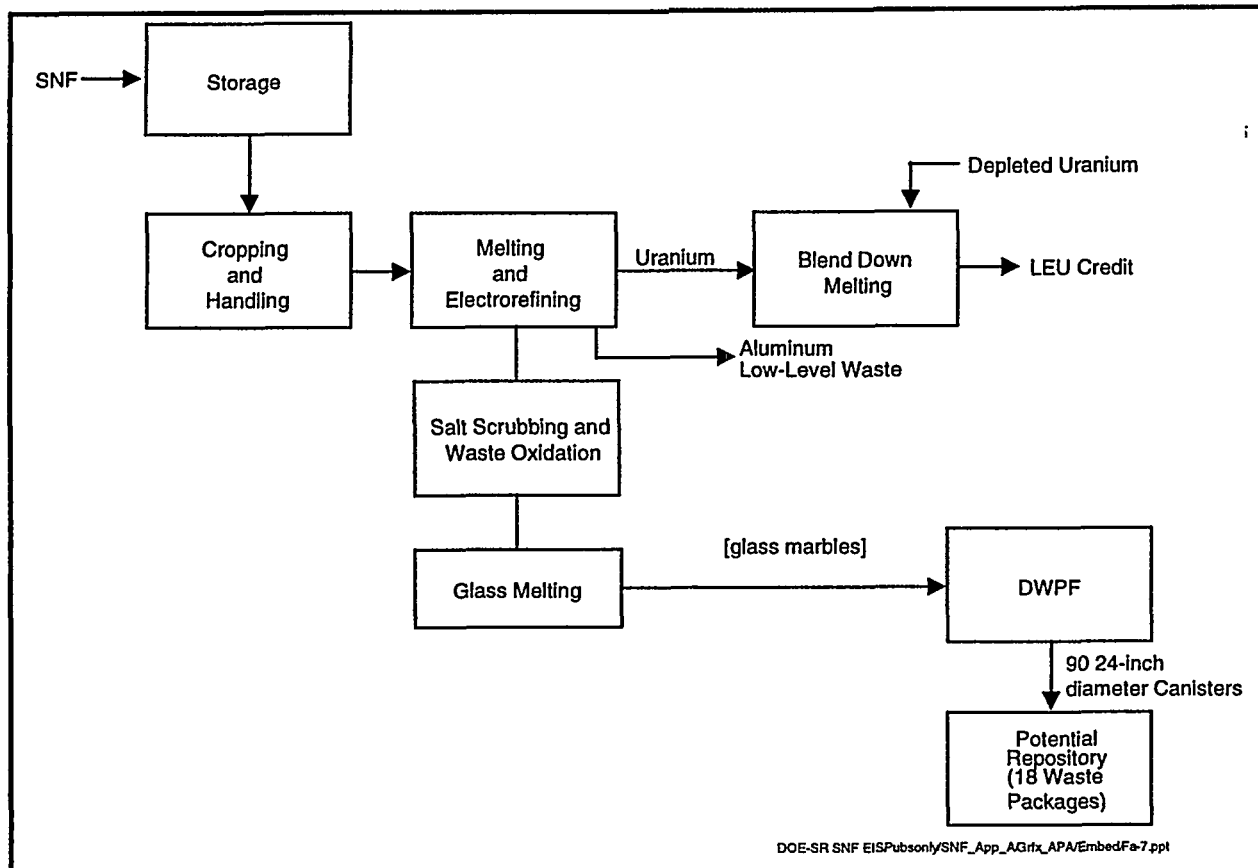
With the continued electrorefining of the SNF, alkaline earth fluorides would build up in the aluminum electrorefiner, and the rare earth and transuranic fluorides would build up in the uranium electrorefiner. These waste products could be separated from the electrolyte by ion exchange or chemical reduction and oxide precipitation.

Waste Treatment

The Electrometallurgical Treatment process would produce several waste streams besides aluminum: scrubbed alkaline earths, rare earths, transuranics, the metal remaining in the anode after uranium electrorefining, and the fibrous aluminosilicate filter used to collect volatile fission products released during SNF melting. These wastes would be placed in an air oxidation furnace to burn to an oxide powder with noble metal fines dispersed in the powder. A small glass melter would melt the oxide powder with glass-forming materials to produce a glass similar to that produced in the Defense Waste Processing Facility. The glass would be formed into marbles for shipment to the Defense Waste Processing Facility for incorporating into high-level waste glass logs. The electrometallurgical treatment would produce about 90 24-inch diameter canisters to be contained in 18 repository disposal packages.

Uranium Dilution

A small melter would melt the uranium metal and blend it with depleted uranium to produce a uranium enriched to about 5 percent uranium-235. This uranium could be sold as feedstock for commercial nuclear fuel manufacture. Figure A-7 shows the Electrometallurgical Treatment process flow diagram.



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Figure A-7. Electrometallurgical Treatment process flow diagram.

A.2.8 TECHNOLOGIES NOT ANALYZED

This section describes technologies that DOE considered but did not analyze further in the EIS because the technologies need further research to demonstrate they are technically viable and cost effective. These technologies have not undergone bench-scale demonstrations.

Chloride Volatility

Chloride volatility is an advanced treatment technology being investigated at the Idaho National Engineering and Environmental Laboratory. The process segregates major nonradiological constituents from SNF for the purpose of volume reduction, and isolates the fissile material to produce a glass or ceramic waste form.

The process is based on completely volatilizing the fuel elements and separating the gaseous

constituents. The fuel would react with chlorine gas at a temperature greater than 1,200°C (2,200°F) to produce volatile chlorides. The fission products and transuranics would be separated by passing the gas through molten zinc chloride in a counter-current scrubber. The gases minus the fission products and transuranics would flow through a series of condensers to remove chloride compounds by fractional distillation. The series of uranium chlorides could be recovered separately, if desired.

The molten zinc chloride would be regenerated by vacuum distillation to recover it for recycle. The fission product and transuranic residue would be converted to oxides or fluorides by fluorination for vitrification and melting with glass frit additives. As an alternative, the residues could be oxidized by boric acid at high temperatures. The transuranics could be separated from the fission products by solvent extraction if separate disposal were necessary.

Although there has been no experimental work with this technology, DOE has determined that the time and expense to overcome the technical risks would be too great. Therefore, this technology is not analyzed further in this EIS.

Can-in-Canister

The Can-in-Canister concept was developed to dispose of excess plutonium. The method would place an array of stainless-steel cans containing plutonium ceramic in a high-level waste canister. The molten high-level waste glass would be poured around the cans. The placement structure would maintain spacing between the cans and the wall of the canister. The Can-in-Canister method is a potentially favorable method for disposing of plutonium because the radiation fields emanating from the high-level waste would discourage intrusion to recover the plutonium. Plutonium itself does not produce high radiation fields.

Can-in-Canister technology is not as attractive for SNF. Most SNF produces high radiation fields that would render recovery difficult and thus would not need the added deterrent of high-level waste surrounding it. In addition, because the melting point of aluminum is less than that of glass in high-level waste vitrification operations, the aluminum fuel could melt in the cans as the canister was being filled with the molten high-level waste glass. Finally, it is not certain the integrity of the glass could be maintained if it contained large voids - in common with the disposal of plutonium in glass.

The Direct Co-Disposal technology provides all the benefits of Can-in-Canister technology without the disadvantages. The SNF would be surrounded by high-level waste glass canisters, ensuring that HLW radiation fields would render the SNF inaccessible for long periods of time. The fuel would not displace any high-level waste canisters, thus eliminating the need for additional repository waste packages. Also, the SNF would not be heated near or over its melting point. For these reasons, the Can-in-Canister process was not analyzed in this EIS.

A.3 Conventional Processing

As discussed in Chapter 2, DOE could use F or H Canyon to process SNF. F Canyon historically has been used to recover depleted uranium and plutonium from depleted uranium target materials irradiated in SRS reactors. H Canyon historically has been used to recover highly-enriched uranium and neptunium from SNF. The following paragraphs are applicable to operations in H Canyon. F Canyon operations would be similar.

At the SNF wet storage basins, the fuel would be placed in aluminum bundle sleeves. For materials test reactor-like fuel elements, the fuel would be stacked four to five elements high in the bundle sleeve. Before shipment to H Canyon, the bundle sleeves would be assembled into larger arrays to make a complete bundle. The size of the array would be determined by shipping cask and the size of the dissolver, and by criticality concerns. Bundling would facilitate handling and maintain a noncritical geometry as the fuel was charged to the dissolver. The storage racks in the Receiving Basin for Offsite Fuel and L-Reactor Disassembly Basin use bundle sleeves to maximize storage space.

The SNF would be transported in a water-filled cask on a rail car from either L-Reactor Disassembly Basin or the Receiving Basin for Offsite Fuel. Inside the airlock doors to the hot canyon, the fuel would be unloaded and placed in an interim wet storage basin to await processing. The bundles of SNF would be fed into the top of a dissolver tank. The fuel would be dissolved in hot nitric acid, producing a solution of highly-enriched uranium, fission products, aluminum, and small amounts of transuranic materials such as neptunium and plutonium.

Head-end processing would use two clarification steps to remove undesirable contaminants that could impede the subsequent solvent extraction process. Gelatin would be added to precipitate silica and other impurities. The

clarified solution would be adjusted with nitric acid and water in preparation for the first-cycle solvent extraction. The waste stream generated from the head-end process would be chemically neutralized and sent to the high-level waste tanks.

The first-cycle solvent extraction in the hot canyon would remove the fission products and other impurities, and then separate the uranium from the other actinides. Nonuranium actinides would not be recovered. If necessary, a second-cycle solvent extraction could further purify the uranium solution. The solvent would be recovered for reuse, the acid solution containing the fission products would be neutralized and transferred to the high-level waste tanks, and the uranium in a uranyl nitrate solution would be transferred to H-Area tanks to be blended down to about 5 percent uranium-235. The uranyl nitrate could be made available for commercial sale.

Chemical processing would generate liquid high-level waste, for which SRS has existing storage and treatment facilities. Impacts associated with the operation of these facilities are described in the Interim Management of Nuclear Materials (IMNM) EIS (DOE 1995) and the Defense Waste Processing Facility (DWPF) Supplemental EIS (DOE 1994). Chapter 5 summarizes the results from the IMNM and DWPF EISs. For completeness, the following paragraphs summarize high-level waste processing at SRS.

Chemical processing produces an acidic solution that is neutralized before transfer to large tanks in the F- and H-Area Tank Farms. During storage to allow short-lived radionuclides to decay, the insoluble components of the alkaline waste settle to the bottom of the tank to form a sludge layer. The liquid supernate is decanted and evaporated to concentrate it into a crystallized salt. Evaporator overheads are condensed and discharged to the F/H Effluent Treatment Facility.

In preparation for final disposal, the salt is re-dissolved and processed to separate it into high-radioactivity and low-radioactivity fractions. The high-radioactivity fraction is sent to the Defense Waste Processing Facility where it is incorporated into a glass form for eventual disposal in a geologic repository. The low-radioactivity fraction is sent to the Saltstone Manufacturing and Disposal Facility where it is mixed with cement, slag, and flyash to produce a cementitious grout solidified in onsite disposal vaults.

The sludge in the high-level waste tanks, after washing to remove dissolved salts, also is transferred to the Defense Waste Processing Facility for incorporation into the high-level waste glass form for repository disposal. About 150 canisters of 24-inch diameter would be produced during the conventional processing of spent nuclear fuel.

Figure A-8 shows the Conventional Processing flow diagram.

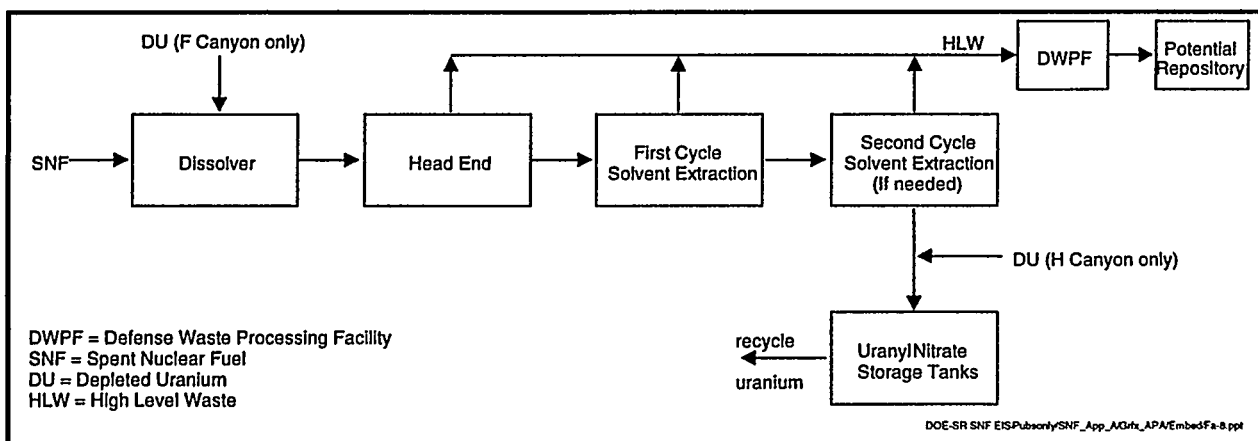


Figure A-8. Conventional Processing flow diagram.

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References

- DOE (U.S. Department of Energy), 1994, *Final Supplemental Environmental Impact Statement, Defense Waste Processing Facility*, DOE/EIS-0082-S, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1995, *Final Environmental Impact Statement, Interim Management of Nuclear Materials*, DOE-EIS-0220, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1996, *Technical Strategy for the Treatment, Packaging, and Disposal of Aluminum-Based Spent Nuclear Fuel, A Report of the Research Reactor Spent Nuclear Fuel Task Team*, Volume 1, Washington, D.C.

APPENDIX B

IDENTIFICATION AND RESOLUTION OF SAVANNAH RIVER SITE SPENT NUCLEAR FUEL VULNERABILITIES

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APPENDIX B. IDENTIFICATION AND RESOLUTION OF SAVANNAH RIVER SITE SPENT NUCLEAR FUEL VULNERABILITIES

B.1 Purpose

The end of the Cold War brought an end to active efforts in the United States to produce nuclear weapons materials such as plutonium. As a consequence, nuclear materials produced for weapons have been stored temporarily for prolonged periods in systems and under conditions not originally designed for long-term storage. Prolonged storage in systems and under conditions designed for short-term storage has degraded the integrity of some of these materials and has led to concerns about safety. These concerns have been documented in reports by both the U.S. Department of Energy (DOE) and the Defense Nuclear Facilities Safety Board (DNFSB). The purpose of this appendix is to provide a compilation of spent nuclear fuel (SNF) storage problems (vulnerabilities) specific to the Savannah River Site (SRS), their recommended corrective actions, and the current status of those corrective actions.

B.2 Introduction and Background

For about 30 years, DOE operated heavy-water reactors at SRS for the production of defense nuclear materials. Low-temperature reactor operation allowed the use of aluminum-clad, aluminum-alloy fuel and aluminum-clad target materials. This reactor design facilitated both fuel and target fabrication and subsequent processing. At the end of a reactor cycle, the fuel and targets were normally discharged to cooling basins and stored for as long as 18 months prior to processing.

In the past, the SRS processed SNF and other reactor-irradiated nuclear materials (RINM) to recover plutonium, tritium, and other isotopes. In April 1992, with chemical separations activities already temporarily suspended, DOE implemented a decision to phase out defense-related chemical separation activities at the SRS. Processing of the "in-process" RINM was

not completed. Facilities designed, constructed, and operated to store RINM for relatively short periods had to store it for relatively long periods pending decisions on the disposition of the materials.

B.3 Spent Fuel Working Group

In August 1993, the Secretary of Energy commissioned a comprehensive baseline assessment of the environmental, safety, and health (ES&H) vulnerabilities associated with the storage of reactor-irradiated nuclear materials in the DOE complex. In October 1993, a multidisciplinary Spent Fuel Working Group, comprised of DOE and contractor employees, assessed 66 facilities on 11 sites. Eight SRS facilities that contained RINM were assessed. The facilities included both wet and dry storage systems. The assessment's objective was to provide an itemized inventory of RINM and an initial assessment of ES&H vulnerabilities associated with the current storage and handling of these materials.

DOE defined vulnerabilities as conditions or weaknesses that could lead to radiation exposure to the public, unnecessary or increased exposure to the workers, or release of radioactive materials to the environment. The loss of institutional controls, such as cessation of facility funding or reductions in facility maintenance and control, could also cause vulnerabilities. Reactor-irradiated nuclear material was defined as spent nuclear fuel and irradiated nuclear targets from production and research reactors; however, it did not include fuel currently in active reactors or irradiated structural materials (other than fuel cladding).

The assessment focused on determining ES&H vulnerabilities and presenting factual information. In general, DOE did not identify or recommend future corrective actions, but did assess corrective actions already under way. Evaluations were made of facilities, structures, systems, operating conditions, and procedures

necessary to protect the workers, the public, and the environment during the storage and in-facility handling of reactor-irradiated nuclear material.

On December 7, 1993, the Working Group released *Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear Fuel and Other Reactor Irradiated Nuclear Materials and Their Environmental, Safety, and Health Vulnerabilities* (DOE 1993) ("The Working Group Report," Volumes I, II and III). Volume I summarized the findings, including: (1) the characteristics and inventory of reactor-irradiated nuclear material; (2) ES&H vulnerabilities associated with different storage options; (3) five generic issues common to many storage facilities; and (4) identification of eight facilities requiring priority management attention, including the SRS L- and K-Reactor Disassembly Basins.

Volume II of the Working Group Report contains Working Group Assessment Team reports for each site, Vulnerability Development forms, and documents used by the Working Group Assessment Team as information sources. Volume II categorized vulnerabilities based on the period during which it was recommended that the vulnerability be addressed. For each of the eight SRS facilities, vulnerabilities were grouped into one of three time periods for management attention: less than 1 year, 1 to 5 years, and more than 5 years.

Volume II identified 21 SRS vulnerabilities. A twenty-second vulnerability was identified later. When DOE reviewed the ES&H vulnerabilities, it determined that two (SRS-2 and SRS-3) were not vulnerabilities and obtained agreement from the working group assessors. Table B-1 lists the SRS vulnerabilities and their assigned priorities.

Fifteen vulnerabilities warranting priority management attention, including one potential vulnerability, were identified for the SRS L-, K-, and P-Reactor Disassembly Basins. Four major

vulnerabilities and one generic vulnerability were identified for the Receiving Basin for Off-site Fuel (RBOF). The *Reactor Division Disassembly Basin Management Plan* (Burke 1993) addressed and provided resolution of the vulnerabilities identified for the reactor disassembly basins and RBOF. A February 21, 1997 memorandum reports on the corrective action closure package for the reactor disassembly basins and RBOF vulnerabilities (Burke 1997).

In February 1994, DOE released the first phase of a three-phased plan to remedy vulnerabilities associated with the storage of spent fuel and irradiated materials. The *Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities, Phase I* (DOE 1994a) described actions that had been completed or for which no major funding or policy issues existed. After the Phase I report was issued, DOE resolved most funding issues associated with SNF vulnerabilities. The Phase II Plan of Action (DOE 1994b), published in April 1994, was the product of follow-on work to the Phase I report.

The Phase III Plan of Action (DOE 1994c), the second update to the original Plan of Action, was issued in October 1994. The Phase III report focused on the resolution of critical policy issues and incorporated stakeholder comments on the original Plan of Action and the first update. Table B-2 lists the Phase I, II, and III corrective action plans and their reported status.

B.4 Defense Nuclear Facilities Safety Board Recommendation 94-1

In May 1994, the DNFSB issued Recommendation 94-1, *Improved Schedule for Remediation in the Defense Nuclear Facilities Complex* (DNFSB 1994). The Board expressed its concern that imminent hazards could arise during the next 2 to 3 years unless problems related to the state of reactor-irradiated nuclear material remaining from the production of nuclear weapons were resolved.

Table B-1. SRS vulnerabilities identified in Spent Nuclear Fuel Working Group Report.

Vulnerability	Priority ^a		
	1	2	3
SRS-01, L-Reactor Disassembly Basin: Potential unmonitored build-up of radionuclide and/or fissile materials in sand filters.	√		
SRS-04, L-Reactor Disassembly Basin: Lack of authorization basis in operating sand filter cleanup system for L-Area Disassembly Basin.	√		
SRS-05, L-Reactor Disassembly Basins: Corrosion of aluminum-clad fuel, targets, and components.			√
SRS-06, L-Reactor Disassembly Basin: Cesium-137 activity level in L-Basin.	√		
SRS-07, L-Reactor Disassembly Basin: Determine if gas bubble release above the bucket storage area is a potential hazard at L-Reactor.	√		
SRS-08, K-, L-, and P-Reactors: Lack of reactor authorization basis.	√		
SRS-09, L-Reactor Disassembly Basin: Corrosion of Mark-31A and B target slugs in K- and L-Disassembly Basins.	√		
SRS-10, P-Reactor Disassembly Basin: Hoist rod corrosion.		√	
SRS-11, K- and L-Reactor Disassembly Basin: Reactor disassembly basin safety analysis envelope.	√		
SRS-12, L-Reactor Disassembly Basin: Inadvertent flooding of L-Reactor Disassembly Basin.	√		
SRS-13, K-Reactor Disassembly Basin: Inadvertent flooding of K-Reactor Disassembly Basin.	√		
SRS-14, P-Reactor Disassembly Basin: Inadvertent flooding of P-Reactor Disassembly Basin.	√		
SRS-15, RBOF, P-, L-, C-, and R-Reactors: Conduct of Operations at reactor facilities and RBOF. (NOTE: RBOF is a less-than-1-year vulnerability.)	√		
SRS-16, RBOF: Inadequate tornado protection at RBOF.		√	
SRS-17, RBOF: Seismic vulnerability of RBOF.		√	
SRS-18, H-Canyon: Seismic vulnerability of H-Canyon.			√
SRS-19, F-Canyon: Seismic vulnerability of F-Canyon.			√
SRS-20, K-, L-, and P-Reactor Disassembly Basins and RBOF: Inadequate leak detection system in the underground water-filled RINM storage basin.		√	
SRS-21, K-, L-, and P-Reactor Disassembly Basins: Inadequate seismic evaluation and potential inadequacies of structures, systems and components to withstand a Design-Basis Earthquake.	√		
SRS-22, R-Area: Potential buried spent nuclear fuel.		√	

- a. **Priority 1:** Vulnerabilities identified by the Working Group as warranting immediate management attention.
Priority 2: Vulnerabilities requiring action within 1 year.
Priority 3: Vulnerabilities requiring action within 1 to 5 years (DOE 1994a).

Table B-2. Status of Savannah River Site Vulnerability Corrective Action Plans.

Identified vulnerabilities	Corrective action plans	Status
SRS-1: Potential unmonitored buildup of radionuclide/fissile material in sand filters. (L-Basin)	SRS-1a: Perform characterization analysis of isotopes in existing sand filter system.	Completed 5/95. Characterization analysis completed.
	SRS-1b: As part of the Basis for Interim Operations development, perform safety analysis for buildup of fissile material in sand filter system and potential for criticality in filters (see items 8 and 11).	Completed 7/96. The criticality safety evaluation determined that there is not an identifiable mechanism by which a critical configuration could be assembled in the disassembly basin sand filter. The L-Basin Basis for Interim Operations (WSRC 1996) concluded that a criticality is not a credible event in the sand filter.
SRS-4: Lack of characterization and updated safety analysis for fissile material in sludge on basin floors and sand filter cleanup systems (K- and P-Basins). (generic issue)	SRS-4a: Complete development and application of technologies required (at L-Basin) for characterization and analysis, removal, and disposal of sludge from L-Basin.	
	SRS-4a.1: Complete characterization of sludge from L-Basin.	Completed 9/93. Sludge analysis completed.
	SRS-4a.2: Complete characterization of sludge from K-Basin.	Completed 6/93. Sludge analysis completed.
	SRS-4a.3: Complete characterization of sludge from P-Basin.	Completed 5/93. Sludge analysis completed.
	SRS-4a.4: Complete removal of sludge from L-Basin.	Phase I (Sludge Consolidation): Completed 3/95.
	NOTE: The ability to maintain excellent basin water quality in the presence of sludge has been demonstrated, eliminating the urgency to consolidate and remove the sludge to prevent further corrosion of stored fuel.	Phase II (Sludge Removal): Completed 1999.
	SRS-4a.5: Complete removal of sludge from K-Basin.	Phase I (Sludge Consolidation): Completed 3/98 (Smith 1998).
	NOTE: The ability to maintain excellent basin water quality in the presence of sludge has been demonstrated, eliminating the urgency to consolidate and remove the sludge to prevent further corrosion of stored fuel.	Phase II (Sludge Removal): Cancelled 4/15/98 (Conway 1998).
	SRS-4a.6: Complete removal of sludge from P-Basin.	Cancelled 4/15/98 (Conway 1998).
	SRS-4b: For characterization and safety analysis of fissile material in sand filters (see item 1).	Completed 7/96. See SRS 1a and 1b.
SRS-6: Cesium-137 activity level in L-Basin water is approaching administrative limits.	SRS-6: Utilize Zeolite in portable ion exchange system to lower Cesium-137 levels in L-, K-, and P-Basins.	Completed 7/96. Zeolite was used in a portable ion exchange system to lower the Cesium-137 levels in L- and K-Basins. P-Basin zeolite cancelled due to fuel consolidation.

TC

Table B-2. (continued).

Identified vulnerabilities	Corrective action plans	Status
SRS-5, 7, and 9: Aluminum-clad fuel and targets are severely corroded, releasing fission products and fissile material to the pool water. Gas bubble release above the bucket storage area at L-Basin might be a potential hazard. (generic issue).	SRS 5,7,9a.01: Modify fuel hangers to provide redundancy against fuel falling to basin floor.	Completed 1/93. No fuel on fuel hangers in P-Basin; fuel in L-Basin Vertical Tube Storage has been removed from hangers and placed in horizontal tube storage; fuel hangers in K-Basin Vertical Tube Storage were inspected and found to be in good condition; the fuel was relocated to horizontal tube storage in July 1997.
	SRS-5,7,9a.02: Develop and implement corrosion surveillance program.	Completed 1/93. The corrosion surveillance program is summarized in <i>Corrosion Surveillance in Spent Fuel Storage Pools</i> (Howell 1996).
	SRS-5,7,9a.03: Complete criticality studies in progress to support the transfer of reactor components from vertical to horizontal storage.	Completed 1/94. Criticality studies to support the transfer of reactor components are documented in <i>100 Area Irradiated Fuel Consolidation and Horizontal Storage Criticality Concerns</i> (Reed 1994).
	SRS-5,7,9a.04: Design and construct racks for horizontal three-deep storage.	Completed 6/95. Horizontal storage rack fabrication and installation completed under Project S-5982 (Guy 1995).
	SRS-5,7,9a.05: Reorient fuel currently stored vertically to the three-deep horizontal array configuration at L-Basin.	Completed 12/95. <i>Storage Solution for Fuel Tubes in the L-Area Vertical Tube Storage</i> (Guy 1993) provided the engineering direction for the reorientation activities. Completion of reorientation is documented in <i>L-3.3 Fuel Bundling</i> (Holmes 1995), and <i>Disassembled Component Log - Fuel Bundling Station</i> (WSRC 1995a).
	SRS-5,7 9a.06: Reorient fuel currently stored vertically to the three-deep horizontal array configuration at K-Basin.	Completed 7/97 (Smith 1998).
	SRS-5,7,9a.07: Modify water chemistry of cleaned basins through the intensive use of portable deionizers (vendor supplied, shock deionization) at L-Basin.	Completed 9/95. Deionization reduced conductivity to 10 µs/cm.
	SRS-5,7,9a.08: Modify water chemistry of cleaned basins through the intensive use of portable deionizers (vendor supplied, shock deionization) at K-Basin.	Completed 1/96. Deionization reduced conductivity to 10 µs/cm.
	SRS-5,7,9a.09: Modify water chemistry of cleaned basins through the intensive use of portable deionizers (vendor supplied, shock deionization) at P-Basin.	Canceled 7/95. Deionization canceled due to P-Basin fuel consolidation (DOE 1995a).
	SRS-5,7,9a.10: Provide deionized makeup water systems for the basins.	Completed 10/95. Systems installed under Project S-5839. Functional performance requirements are documented in <i>Disassembly Basins Upgrades</i> (WSRC 1995b).

Table B-2. (continued).

Identified vulnerabilities	Corrective action plans	Status
	SRS-5,7,9a.11: Assess the hazard of gas releases as a result of the corroding material in the bucket storage area at L-Basin.	Completed 8/95. Evaluation of gas releases study indicated the exposure potential from leaking targets in the basin is insignificant (Hochel 1995).
	SRS-5,7,9a.12: Maintain basin water chemistry through the application of additional dedicated and upgraded deionizers and regeneration capabilities.	Completed 6/96. Continuous deionization systems were installed and tested for K- and L-Basins under Project S-5839 (New 1996a).
	SRS-5,7,9a.13: Assess deionizer regeneration at RBOF facilities to support timely regeneration of L-, K-, and P-Basin ion exchange resins.	Completed 10/94. Assessment of deionizer regeneration facilities at RBOF was documented in <i>Division Critical Item - RBOF Regeneration System Improvements</i> (Cederdhal and Freeman 1994).
	SRS-5,7,9a.14: Complete modifications to regeneration equipment at RBOF if determined appropriate by assessment.	Completed 6/98 (Smith 1998).
	SRS-5,7,9a.15: Complete placement of MK-31 slugs stored in L-Basin into containment boxes to minimize the spread of fission and corrosion products.	Completed 1/94. Containment program was developed and implemented to reduce the spread of contamination from the corroding target slugs. Subsequently, all targets were removed from L-Basin and processed.
	SRS-5,7,9b.1: Develop acceptance criteria and validated heat transfer models for highly enriched uranium aluminum-clad fuel.	Completed 3/96. Acceptance criteria established and documented (Sindelar et al. 1996).
	SRS-5,7,9b.2: Complete development of generic dry storage procurement specification.	Completed 12/95. Specifications completed and documented (New 1995).
	SRS-5,7,9b.3: Complete preconceptual design studies for dry storage option.	Completed 3/12/98 (G-CDR-L-00001) (Smith 1998).
	SRS-5,7,9b.4: Complete Environmental Impact Statement (EIS) for dry storage.	Ongoing, scheduled for 04/00. Notice of Intent issued by DOE to prepare an EIS on management of aluminum-clad fuel at SRS (61 FR 69085). The scope of the EIS will include an assessment of the impacts associated with construction and operation of a dry storage facility. TC
	SRS-5,7,9b.5: Complete civil structural design for dry storage.	Preliminary design work is scheduled for FY03, followed by final design in FY04 and FY05. TC

Table B-2. (continued).

Identified vulnerabilities	Corrective action plans	Status
SRS-8 and 11: Lack of adequate authorization bases (and operating procedures) including updated and approved SAR that addresses long-term storage of RINM and accident mitigation.	SRS 8,11: Complete Basis for Interim Operation (BIO) per DOE 5480.23 and Facility Hazards Assessment per DOE 5500.3A, currently under development for K-Basin (anticipated to be bounding for L- and P-Basins). BIO development will include: - Performance of safety analysis for buildup of fissile material in sand filter system and evaluation of potential for criticality in filters. - Evaluation of basin ventilation requirements. - Evaluation of potential cask drop hazards. - Evaluation of potential hazards associated with basin flooding. - Evaluation of capability to maintain cooling water during seismic events and other credible initiating events. Issues include a loss of emergency cooling source and results of percolation studies.	Completed 9/95. Completed the BIO per DOE Order 5480.23 for K-Basin, bounding for L- and P-Basins, and forwarded to DOE for review in 8/94; DOE approved the BIO in 3/95 (WSRC 1995c). Completed the Facility Hazards Assessment per DOE Order 5500.3A for K-Basin, bounding for L- and P-Basins (WSRC 1995d).
SRS-10: Failure of severely corroded hoist rod bolts on the twin hoist in the P-Basin could result in a fuel drop and subsequent fission product release or violation of criticality safety spacing requirements.	SRS-10: Conduct hoist assembly load test based on revised preventative maintenance procedure. This does not pose a criticality concern. Administrative controls and geometric constraints dismiss the risk of criticality. The hoist assembly load test will be conducted prior to use of the hoist. No use of the hoist is contemplated for the foreseeable future and it is not cost-effective to perform the load test until use of the hoist is required.	Completed 11/95. The Twin Hook Hoist was replaced in P-Area and the replacement hoist load tested using Work Request No. BHBRs (WSRC 1995e).
SRS-12-14: Flooding of basins initiated by human error or seismic event (affecting the makeup system) could result in basin overflow with resultant fission product release to environment.	SRS-12-14: Evaluate potential hazards associated with basin flooding as part of the Basis for Interim Operation development. (See Item SRS 8,11).	Completed 3/95. The potential hazards associated with basin flooding were evaluated as part of the BIO development, which concluded that 105-K, -L and -P buildings are not subject to flooding (WSRC 1995c).
SRS-15: Conduct of operations emphasis on the extended role of SRS fuel storage basins is necessary. (Vulnerability applicable to RBOF and P-, K-, L-, C-, and R-Reactors).	SRS-15: Conduct training to emphasize extended storage of production reactor fuels and target materials.	Completed 2/94. Training specifically designed to address the concerns with extended storage of SNF was conducted. Individual training records are maintained in Building 704-24K.

Table B-2. (continued).

Identified vulnerabilities	Corrective action plans	Status
SRS-16: The roof over the cask basins and the transite walls of the RBOF facility provide inadequate tornado missile protection.	SRS-16: Complete a detailed structural assessment for design-basis hazards (seismic, tornado, etc.). See Item SRS-17 actions. A detailed structural assessment for the design-basis hazards for the facility will be part of the safety analysis report upgrade. RBOF Technical Safety Requirements (TSRs) to be submitted to DOE in FY95. The new Safety Analysis Report will be complete in FY96.	Completed. A detailed structural assessment for the design basis hazards for the facility (seismic, tornado, etc.) has been completed as part of the safety analysis upgrade. The analysis includes the basins, the above-grade walls, roof, and the storage racks. Following the analysis, limited modifications were performed. The analysis has shown RBOF meets DOE requirements for natural phenomenon hazards. (New 1996b). The Safety Analysis Report and TSRs have been approved by DOE.
SRS-17: Since the initial design, there has been no deterministic seismic evaluation of the facility. A seismic event could damage masonry walls above the pool, the sole significant makeup water line penetrating the facility foundation, and unanchored chemical storage tanks/piping adjacent to the facility, adversely affecting the RBOF. (Storage racks, although anchored to the floor and wall of the basin, are not seismically qualified.)	SRS-17: Complete a detailed assessment in conjunction with the efforts to upgrade the Safety Analysis Report. A detailed structural assessment for the design-basis hazards for the facility (seismic, tornado, etc.) will be part of the safety analysis report upgrade. The analysis will include the basins, the above-grade walls, roof and storage racks. In the interim, the existing safety analysis bounds radiological releases from all credible scenarios because it is extremely conservative. The new Safety Analysis Report will be complete in FY96.	Completed. Actions accomplished in response to SRS-16 have also completed the corrective actions required for SRS-17. (See Issue SRS-16.) The Safety Analysis Report was developed and submitted for DOE-SR review and approval in August 1997 (Smith 1998). Safety Analysis Report and TSRs have been approved by DOE.
SRS-18: Potential seismic vulnerability of H-Canyon because the facility does not meet current seismic design standards. Initial seismic calculations indicate that portions of the H-Canyon facility that house the target storage vault are not structurally adequate, which could result in a direct release path.	SRS-18a: Complete detailed seismic structural assessment along with Safety Analysis Report upgrade. SRS-18b: Complete development of new Technical Safety Requirements.	Completed. Analysis has shown H-Canyon meets DOE requirements of seismic resistance (Alm 1997). Also, a recent review of U.S. Geological Survey hazard maps indicates the map results would have no significant impacts on hazard results used for H-Canyon seismic analysis work (WSRC 1997a). Completed. Technical Safety Requirements were approved 4/97.
SRS-19: Potential seismic vulnerability of F-Canyon because the facility does not meet current seismic design standards. Initial seismic calculations indicate that portions of the F-Canyon facility that house the fuel storage vault are not structurally adequate, and that could result in a direct release path from the facility in the event of a Design Basis Earthquake. A criticality could result from seismically-induced damage to the storage racks, which could result in additional radiation release.	SRS-19: Complete detailed seismic analysis for F-Canyon.	Completed 8/96. The detailed seismic calculations were completed in 7/96. DOE-SR sent recommendations to DOE Headquarters to resume operations because results of seismic calculations were favorable. DOE Headquarters approved resumption of F-Canyon operations, which started on 8/26/96 (Alm 1996a). Also, a recent review of U.S. Geological Survey hazard maps indicates the map results would have no significant impacts on hazard results for F-Canyon seismic analysis work (WSRC 1997a).

Table B-2. (continued).

Identified vulnerabilities	Corrective action plans	Status
SRS-20: Lack of adequate leak detection system for the storage basins. The current leak detection method is not sufficiently sensitive to detect small leaks.	SRS-20a (for L-, K-, and P-Basins): Install additional monitoring wells (two per basin) for L-, K-, and P-Basins to ensure detection of basin leakage.	Completed 3/95. Monitoring wells for L-, K-, and P-Basins completed 3/2/95, (Burbage 1995). Installation completed under Project S-5839 (New 1996a).
	SRS-20a (for RBOF): Perform studies of other beneficial methods of leak detection.	Completed 1/95. Overall H-Area groundwater effects are monitored and reported in accordance with applicable requirements (Clark 1994; Burbage 1995).
	SRS-20b (for L-, K-, and P-Basins): Improve the basin leak detection threshold by comparing trends in radionuclide concentration observed in the monitoring wells and the basins. Chemical constituents of samples from the monitoring wells and the basins will be monitored and trended for comparison purposes.	Completed 12/96. Basin level trending capabilities improved with the installation of upgraded simple level monitoring instruments and a makeup water system flow totalizer. Engineering initiated basin level trending, which provides more accurate monitoring of changes in the basin level. Monitoring wells downgradient of the basins have improved the dispersion/dilution models. Comparison of monitoring well chemical constituent trend data to basin water data has been initiated; monitoring wells are sampled monthly; evaluations of the radionuclide concentrations are issued in a quarterly report (Burbage 1996).
	SRS-20b (for RBOF): Evaluate and if necessary install improved level detection.	Completed 1/95. A study of the beneficial methods of leak detection included a review of the level monitoring capability used at the West Valley facility in New York. No significant benefits from implementation of this system were identified. Overall H-Area groundwater effects are being monitored and reported in accordance with applicable requirements. The RBOF level detection has been determined to be adequate (Clark 1994; Burbage 1995).
	SRS-20c (for L-, K-, and P-Basins): Evaluate the need for improved level detection system to provide more accurate monitoring of changes in basin level. (The accuracy of the current basin level indication is within 7,570 liters [2,000 gallons]).	Completed 12/94. Several options for level and leak detection systems were evaluated. DOE determined that the installation of an upgraded, simple level monitoring instrument coupled with the new makeup water and monitoring well system will provide an adequate cost-effective basin water inventory information system (New 1994).

Table B-2. (continued).

Identified vulnerabilities	Corrective action plans	Status
SRS-21: Inadequate seismic evaluation and potential inadequacies of structures, systems, and components to withstand a Design-Basis Event. The potential exists for: the failure of basin expansion joints and water stops, causing a release of radioactive materials to the environment; the failure of vertical tube storage frames or a load drop onto fuel assemblies causing damage or reconfiguration of fuel and possible criticality, due to a seismic event. (generic issue)	SRS-21a: Complete soil stability assessment for input to seismic analyses for L-, K-, and P-Basin that is in progress. SRS-21b: Complete seismic evaluations if determined to be necessary as a result of the Basis for Interim Operations development, or soil stability assessments for L-, K-, and P-Basin. A recent assessment of the K-Basin exterior walls and foundations determined they could withstand a 0.2g earthquake. Minor leakage could occur but would be slow. A recent assessment determined that the consequences of an earthquake for L- and P- Basins are less than those for K-Basin because K-Basin has the highest radionuclide inventory. SRS-21c: Depending on results of BIO completed by 11/94, development of accident mitigation procedures might be appropriate. SRS-21d: See Item 5,7,9a for reorientation of fuel.	Completed 10/94. Soil stability assessments for K- and P-Basins are not required (Burke 1994). Geotechnical investigation into L-Areas is ongoing. Completed 3/95. Soil stability assessments for K-, L-, and P-Basins are not required (Burke 1994). These assessments support seismic analyses of emergency cooling systems; however, the BIO (WSRC 1995c) determined that fuels stored in the basins do not require emergency cooling after a postulated Design-Basis Event resulting in basin draindown.
SRS-22: Potential vulnerability in buried fuel at SRS.	SRS-22: Fuel failure of a Mark V fuel assembly occurred in the R-Area disassembly basin in 1957. Over 7 years, the fissile materials in the failed fuel assembly completely oxidized. In 1964, the remains of the assembly were retrieved with no appreciable amount of fuel or fission products remaining in the assembly. All of the oxidized fuel near the assembly was removed using filters and deionizers and subsequently processed in RBOF. The fuel material is currently held in authorized basins and tanks. No further action is required.	Completed 7/96. Accident mitigation procedures are not required because this accident would result in low consequences. Completed 7/97. Completed 12/93.
NOTE: This item was discussed in the Spent Fuel Working Group Report summary but not addressed in vulnerability development forms. Disassembly basins do not have high efficiency (confinement/negative pressure) ventilation systems.	NOTE: The K-Area Basis for Interim Operation addresses the existing facility and safety margins with respect to the need for airborne release containment. The results of the BIO show there is adequate safety margin without facility upgrades. Upgrades are not required for several reasons: - Over 40 years of operating experience with no events involving the spread of particulate radioactive contamination from the basins. - Little stored energy in the disassembly basins because of radioactive decay. - Radiation level in the disassembly basin area is low (<2 millirem/hour) and the contribution from airborne particulate matter is negligible.	Completed 3/95. Area BIO was approved by DOE; no further action is required (WSRC 1995c).

DNFSB 94-1 addressed vulnerabilities at several DOE sites, including the following SRS vulnerabilities concerning SNF and related solutions, tanks, and processing activities:

Several large tanks in the F-Canyon at the Savannah River Site contain tens of thousands of gallons of solutions of plutonium and trans-plutonium isotopes. These tanks, their appendages, and vital support systems are old, subject to deterioration, prone to leakage, and they are not seismically qualified.

Processing canyons and reactor basins at the Savannah River Site contain large amounts of deteriorating irradiated reactor fuel stored under conditions similar to those at the 603 Basin at INEL [Idaho National Engineering Laboratory].

There are thousands of containers of plutonium-bearing liquids and solids at ... SRS Large quantities of plutonium solutions are stored in deteriorating tanks, piping, and plastic bottles.... It is well known that plutonium in contact with plastic can cause formation of hydrogen gas and pyrophoric plutonium compounds leading to a high probability of plutonium fires.

The slow pace of remediation and additional delays in stabilizing materials might be accompanied by further deterioration of safety and unnecessary increased risks to workers and the public.

DOE accepted the Board's Recommendation on August 31, 1994, and issued *The Defense Nuclear Facilities Safety Board Recommendation 94-1 Implementation Plan* (DOE 1995b). Table B-3 summarizes the SRS vulnerabilities identified in DNFSB 94-1 and the associated commitments made by DOE in the Implementation Plan.

B.5 DNFSB January 1995 SRS Spent Fuel Vulnerability Assessment

In January 1995, members of the DNFSB staff assessed SRS progress toward resolving vulnerabilities associated with the storage of spent fuel. Their *SRS Spent Fuel Storage Trip Report*, January 23, 1995 (Burnfield 1995) stated that although DOE Headquarters had agreed to approach spent fuel vulnerability problems using a systems approach, they had been slow in implementing that approach and there was little evidence of SRS applying the approach to the spent fuel management project.

DNFSB members expressed concern that aggressive action was not being taken to resolve vulnerabilities related to improving the water chemistry of the basins. They also highlighted two new areas of concern related to RBOF.

The report acknowledged that Westinghouse Savannah River Company had initiated an aggressive program to ensure that the risks associated with these two new areas of concern were accurately quantified and were acceptable. However, the efforts for the two areas were not tied together and therefore could result in an inability to link the two hazards successfully.

The DNFSB did not formally submit these issues and the trip report to DOE. As a consequence, formal corrective actions were not developed nor were these issues entered into and tracked by the DOE Safety Issues Management System. However, Westinghouse Savannah River Company performed two nuclear criticality safety evaluations to address these issues: *Reactivity Effects of Tilting Fuel Assemblies and Bundles in RBOF* (Reed 1995) and *Credible Water Depth for Criticality Incidents*

Table B-3. Applicability of Defense Nuclear Facilities Safety Board Recommendation 94-1 to SRS.

Identified vulnerabilities	Implementation plan commitment	Status	
Sub-recommendation (3): That preparation be expedited to process dissolved plutonium and transplutonium isotopes in tanks in F-Canyon at the Savannah River Site into forms safer for interim storage. The Board considers this problem to be especially urgent.	A stabilization method for F-Canyon has been selected. Stabilization of plutonium solutions began in February 1995 and will be completed by January 1996. A conceptual design report for the stabilization of americium and curium solutions will be completed by December 1995. All americium and curium solutions will be stabilized by September 1998. Other solutions not specifically mentioned in this recommendation but addressed in this plan will be stabilized in accordance with the following schedule: • Plutonium-242 solution in H-Canyon by November 1997 • Highly enriched uranium solutions at SRS by December 1997 • Plutonium-239 solution in H-Canyon by February 2000 • Neptunium solutions in H-Canyon by December 2002	Stabilization of plutonium solution was completed in 4/96. The Conceptual Design Report for the stabilization of americium and curium solutions was completed in 11/95. The current schedule for americium/curium solution stabilization calls for completion by 9/02. This schedule may be rebaselined in 4/00. Stabilization of Plutonium-242 in H-Canyon was completed in 12/96. Highly enriched uranium solutions continue to be stored safely. The schedule for disposition of H-Canyon uranium solutions calls for stabilization by 12/03. This schedule may be rebaselined in 4/00. Stabilization of H-Canyon plutonium-239 solutions is forecast for completion in 2002, and stabilization of H-Canyon neptunium solutions is forecast for completion in 12/05. These schedules may be rebaselined in 4/00.	TC
Sub-recommendation (5): That preparation be expedited to process the containers of possibly unstable residues at the Rocky Flats Plant and to convert constituent plutonium to a form suitable for safe interim storage.	... Residues at other sites, not specifically addressed in this recommendation will be stabilized according to the following schedules: • Sand, slag, and crucibles at SRS by December 1997	Stabilization of sand, slag, and crucible at SRS began 10/97 and is forecast for completion by 7/98.	
Sub-recommendation (6): That preparations be expedited to process the deteriorating irradiated reactor fuel stored in basins at SRS into a form suitable for safe storage until an option for ultimate disposition is selected.	The method for stabilizing fuel and targets at SRS will be selected by July 1995 pursuant to the Interim Management of Nuclear Materials (IMNM) EIS and ROD. Fuel storage basin water chemistry upgrades will be completed by May 1996. Contingent on the outcome of the IMNM EIS, targets will be stabilized via dissolution by September 1996; fuel dissolution will be completed by November 1999. Stabilization of resultant uranium solutions will be completed by April 2000.	Stabilization of Mark 31 targets was completed in 1/97. Stabilization of Mark 16 and 22 fuel assemblies began 7/97 and is forecast for completion in 2001. Fuel storage basin water chemistry upgrades were completed in 5/96. HEU from fuel will be blended down to LEU on a schedule that supports transfer of the LEU to commercial industry.	TC
Sub-recommendation (8): That those facilities that may be needed for future handling and treatment of the materials in question be maintained in a usable state. Candidate facilities include, among others, F- and H-Canyons and FB- and HB-Lines at SRS,	Sufficient capabilities will be retained to maintain future handling, treatment and safe storage of the materials addressed in this plan. A discussion of facilities currently in use or planned for use is included in Section 2.6. The facilities section of the Integrated Program Plan will be prepared by December 1995.	The Integrated Facilities Plan (DOE 1995a) addressed the utilization of the F- and H- Canyons.	
Sub-recommendation (9): Expedited preparation to accomplish actions in items (3) through (8) above should take into account the need to meet the requirements for operational readiness in accordance with DOE Order 5480.31.	Facilities will be started or restarted in accordance with DOE Order 5480.31. These restart and startup requirements will be taken into account in the development of the facilities.	Operational Readiness Reviews for restart of the following facilities have taken into account requirements of DOE Order 5480.31: FB-Line (complete 11/95); F-Canyon (complete 9/26); H-Canyon Dissolving (complete 7/97); HB-Line Dissolving (complete 3/98)	

in RBOF (Reed 1996). Table B-4 summarizes the SRS vulnerabilities identified in the trip report and results of the safety evaluations.

B.6 DNFSB June 1995 SRS Spent Fuel Vulnerability Assessment

In June 1995, members of the DNFSB visited the SRS to review SNF activities related to the implementation of DNFSB Recommendation 94-1 (see Section B.4). The DNFSB Chairman formally transmitted the report to DOE and identified issues that were "... not being adequately considered in the evaluation of remediation alternatives." In addition, the report identified badly corroding foreign fuel in RBOF that DOE had categorized as stable (Conway 1995).

The corroding foreign fuel that DOE had categorized as stable was failed Taiwanese Research Reactor Fuel and Experimental Breeder Reactor slugs that were being stored in cans in RBOF. Although the damaged fuel was housed in protective cans, the cans were leaking and continued deterioration was likely. DOE responded to

this concern by issuing the Interim Management of Nuclear Material EIS Record of Decision (60 FR 65300), which identified chemical processing as the preferred alternative for the at-risk foreign fuel.

In the *Draft Environmental Impact Statement, Interim Management of Nuclear Materials* (DOE 1995c), DOE identified Processing to Metal as the preferred alternative for the remediation of Mark-16 and -22 fuels. DNFSB became aware that DOE was considering dry storage of aluminum-clad fuel as an alternative to chemical processing. DOE issued the Final IMNM EIS (DOE 1995d) with No Action as the preferred alternative for these fuels to allow time for further consideration of dry storage. DNFSB expressed three specific concerns on the stabilization technologies for Mark-16 and -22 fuel. DOE's final decision was to identify Chemical Processing as the preferred alternative for this material, as recorded in the second IMNM Record of Decision (61 FR 6633).

Table B-5 summarizes the SRS vulnerabilities identified in the trip report.

Table B-4. Vulnerabilities identified in the January 1995 Defense Nuclear Facilities Safety Board Trip Report.

Identified vulnerabilities	Implementation plan commitment	Status
Some fuel in RBOF is stored vertically in racks, allowing the fuel to lean from top to bottom slightly, resulting in a violation of criticality safety requirements.	The DNFSB did not choose to submit these issues and this trip report formally to DOE. Consequently, formal corrective actions were not developed nor were these issues entered into and tracked by the DOE Safety Issues Management System.	Calculations documented in <i>Nuclear Criticality Safety Evaluation: Reactivity Effects of Tilting Fuel Assemblies and Bundles in RBOF</i> (Reed 1995) resulted in the conclusion that, "There are no current situations in RBOF Storage Basin #1 in which the configuration has been determined to be more reactive than a k_{eff} of 0.95, the Technical Standard limit."
The amount of water shielding was misidentified in the safety documentation.	The DNFSB did not choose to submit these issues and this trip report formally to DOE. Consequently, formal corrective actions were not developed nor were these issues entered into and tracked by the DOE Safety Issues Management System.	An analysis documented in <i>Nuclear Criticality Safety Evaluation: Credible Water Depth for Criticality Incidents in RBOF</i> (Reed 1996), concluded that, "Thus, there is no basis to define a NIM evacuation zone (region in which personnel can receive 12 rads or more as a result of a criticality incident) for the RBOF basins. Based on requirements of DOE Order 5480.24, a criticality alarm system is not required for the RBOF basins."

Table B-5. Vulnerabilities identified in the June 1995 Defense Nuclear Facilities Safety Board Trip Report.

Identified vulnerabilities	Implementation plan commitment	Status
1. Contrary to the Implementation Plan for Recommendation 94-1, it appears that dry storage is being considered as the preferred alternative to remediate Mark-16 and -22 fuel assemblies at the SRS. Although dry storage, as well as chemical separation, can achieve stable conditions, the following concerns could affect the decision to dry store this deteriorating fuel:		
1.a The requirements for dry storage of highly enriched aluminum-clad spent nuclear fuel have not been developed. This represents a large uncertainty in the time and effort required to achieve dry storage and a large uncertainty in the time during which continued wet storage will be required.	Formal corrective actions were not developed nor were these issues entered in and tracked by the DOE Safety Issues Management System.	The preferred alternative for remediation of these fuels was Blending Down to Low-Enriched Uranium, as recorded in the second IMNM Record of Decision (61 FR 6633).
1.b The waste acceptance criteria needed to transition dry-stored aluminum-clad spent nuclear fuel to a geologic repository have not been developed. This raises the possibility of having to rehandle, repackage, or even process this material in the future to meet storage requirements.	Formal corrective actions were not developed nor were these issues entered in and tracked by the DOE Safety Issues Management System.	The preferred alternative for remediation of these fuels was Blending Down to Low-Enriched Uranium, as recorded in the second IMNM Record of Decision (61 FR 6633).
1.c Lengthy delays needed to implement dry storage will extend by years the period of wet storage of the deteriorating spent fuel and allow continued corrosion. This will aggravate the problems of continued degradation, potential environmental insult, radiation exposure, and waste generation.	Formal corrective actions were not developed nor were these issues entered in and tracked by the DOE Safety Issues Management System.	DOE responded to these concerns by changing the preferred alternative for remediation of these fuels to Blending Down to Low-Enriched Uranium, as recorded in the second IMNM Record of Decision (61 FR 6633).
2. In addition, corroding spent fuel in RBOF is releasing more than twice the amount of fission products to the basin water than the corroding Mark-31 targets are releasing to the L-Basin. This significant corrosion is contaminating the facility, generating significant waste, and contributing to personnel exposure. Surprisingly, DOE plans to keep the current inventory of fuel at RBOF in wet storage for the next 10 years. A more urgent response is merited.	Formal corrective actions were not developed nor were these issues entered in and tracked by the DOE Safety Issues Management System.	DOE elected to stabilize all fuel in RBOF with potential leakage. (First Record of Decision [60 FR 65300] and fourth Record of Decision [62 FR 17790].)

B.7 DNFSB 1996 SRS Spent Fuel Handling Assessment

In August 1996, members of the DNFSB assessed SRS spent fuel handling and processing operations. The Board noted that the transfer of spent fuel requires moving massive casks in spent fuel storage basins where a cask drop could cause structural damage and significant water inventory loss. Their Trip Report (Conway 1996) reported the following concerns associated with stabilization operations and the retrieval of spent fuel from the K, L, and P Basins:

1. There is no assurance that makeup water will be available after a design-basis accident.
2. The crane rope is corroded, and the fatigue life of some cranes is not known.

3. A qualified rigger is not present during critical cask lifts.
4. Although fuel is being removed from the basins, significant quantities of activated scrap metal will remain.

DOE first responded to these concerns in a letter to the Board dated November 21, 1996 (Alm 1996b). DOE provided a more detailed response in a letter dated December 13, 1996 (Alm 1996c). Table B-6 summarizes the SRS SNF transfer vulnerabilities identified in the 1996 Trip Report and the associated DOE responses and commitments (Potvin 1997).

Table B-6. Vulnerabilities identified in the August 1996 Defense Nuclear Facilities Safety Board Trip Report.

Identified vulnerabilities	Implementation plan commitment	Status
There is no assurance that makeup water will be available if a cask drop or seismic event should cause a leak. Basin water is supposed to be replaced by raw untreated water from the Emergency Service Water system; however, this line is not tested regularly, it has not been used for more than a year, and it is not seismically qualified.	Studies indicate that dropping the 63.5 metric-ton (70-ton) cask, as analyzed in the recently issued Basis for Interim Operation for L-Reactor, could potentially result in a crack with a maximum leak of 397 liters (105 gallons) per minute. At this rate, operators would have a minimum of 6 days to implement mitigative actions before radiation levels began to increase, at which point all workers in the vicinity would be evacuated. In such an event, operators could respond by implementing various procedures using available systems to restore basin water levels.	A letter to the DOE-SR DNFSB Liaison (Voss 1997a) identified plans and procedures developed in response to this vulnerability.
A qualified rigger is not present during fuel cask lifts. Fuel cask lifts are critical and preengineered. However, a crane operator, who has only Incidental Rigger Training, performs both the rigging and crane movement. This seems to contradict the SRS Hoisting and Rigging Manual, which states that a rigger shall ensure (1) the rigging equipment has the required capacity and is in good condition, (2) the rigging equipment is per procedure, and (3) the load path is clear.	Although detailed preplanned and demonstrated emergency capabilities are not required for accident scenarios that would allow adequate time for facility workers to respond, an integrated facility response to a basin leak is being developed. This response, to be completed in the second quarter of FY 97, will consist of a combination of operational procedures and engineering response plans with the objective of mitigating basin leakage.	Preengineered lifts are established for routine, repetitive lifting jobs such as cask movement. For these lifts, established procedures define the rigging equipment and the process used. These cask-handling procedures are reviewed by fully qualified Site rigging personnel.
Corrosion is evident along the entire length of the K-Basin cask crane's wire rope and the fatigue life of basin cranes is not known. DNFSB staff were not able to view the L-Basin crane rope, but were told its condition is similar. ASME B30.2-1990 identifies excessive corrosion on wire rope as a hazard. WSRC stated that the rope is adequate based on visual inspection by site riggers. However, as noted in the Construction Safety Association of Ontario's Rigging Manual, visual inspection gives a poor indication of the extent of degradation since corrosion often begins inside the rope. A more rigorous inspection includes examining the rope	Facility operators attend "incidental rigger/operator" training. The "incidental rigger" is trained to ensure the rigging equipment has the required capacity and is in good condition, the rigging equipment is utilized per procedures, and the load path is clear. These qualifications are appropriate for the routine preengineered operations conducted in the facility.	This commitment is complete (WSRC 1997b).
	SRS has a comprehensive crane inspection program, which is based on a compilation of various national and international codes and standards. The quarterly and annual crane inspections are performed in accordance with <i>Overhead and Gantry Cranes</i> , ASME B30.2, Chapter 2-2. The wire ropes are inspected on a monthly frequency pursuant to "Overhead and Gantry Cranes," 29 CFR 1910.179(m), and inspection criteria in accordance with ASME B30.2, Section 2-2.4.	DOE responded to this as documented in a letter (Sidey 1997). On January 20, 1997, additional references were provided in response to Commitment 2 (Voss 1997b).
	Crane inspection program documents for the K- and L-Basins and the Receiving Basin for Off-site Fuel to be provided to the DNFSB will describe inspection requirements, frequency and criteria, load tests, dye-penetrant tests, etc.	

Table B-6. (continued).

Identified vulnerabilities	Implementation plan commitment	Status
core. If the core is corroded, the cranes' safety factor may be much less than WSRC believes.	As part of scoping stage for determining the need for a crane upgrade proposal for DOE, a representative from the crane manufacturer will perform a baseline inspection/evaluation of the 11-metric-ton (85-ton) crane and issue a report. All documentation will be transmitted within 60 days of the crane manufacturer's inspection.	
Although fuel is being removed from the basins, significant quantities of activated scrap metal will remain. Besides fuel, the basins store buckets of highly radioactive scrap metal. These buckets are suspended by rope and corroding wire cables. In contrast with the fuel, no plan of action has been formulated for retrieval of this material.	As part of efforts to enhance water quality and reduce hazards, WSRC has been pursuing removal of selected materials from the basins. Activities to date have focused on removal of the more serious hazards, including cadmium control rods, corroded fuel, and excess radioactive sources. Future efforts will be directed at the remaining materials such as irradiated metal and contaminated scrap. DOE has recently improved methods for disposing of waste at SRS. These are part of a waste certification program that ensures the identification of characteristics of all wastes to enable proper disposal. The RBOF Waste Certification Plan is in revision and will reflect the new waste disposal process. Implementation schedules will be based on resource and budget availability.	After discussions with DOE, it was determined that no deliverable was required to be forwarded to DNFSB concerning this issue.

References

- Alm, A. L., 1996a, "F-Canyon Nuclear Material Stabilization Activities," internal memorandum to M. J. Fiori, U.S. Department of Energy, Washington, D.C., August 20.
- Alm, A. L., 1996b, Letter to John T. Conway, (Chairman, Defense Nuclear Facilities Safety Board), U.S. Department of Energy, Washington, D.C., November 21.
- Alm, A. L., 1996c, Letter to John T. Conway, (Chairman, Defense Nuclear Facilities Safety Board), U.S. Department of Energy, Washington, D.C., December 13.
- Alm, A. L., 1997, "H-Canyon Seismic Issue and Nuclear Material Stabilization Activities," internal memorandum to M. J. Fiori, U.S. Department of Energy, Washington, D.C., January 27.
- Burbage, G. L., 1995, "Leak Detection at the Receiving Basin for Offsite Fuels by Groundwater Monitoring," interoffice memorandum, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina, January 10.
- Burbage, G. L., 1996, *October 1996 Disassembly Basin Monitoring Data Summary (U)*, SFS-RPS-960028, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Burke, S. D., 1993, *Reactor Division Disassembly Basin Management Plan (U)*, RRD-RSE-93-0075, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Burke, S. D., 1994, *DOE SNF Vulnerability Assessment Action Plan Items 21a and 21b (U)*, RRD-RSE-940284, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Burke, S. D., 1997, internal memorandum, "DOE Vulnerability Assessment Corrective Action Plans Closure (U)," SFS-RSE-960023, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina, February 21.
- Burnfield, D., 1995, *SRS Spent Fuel Storage Trip Report, January 3-6, 1995*, Defense Nuclear Facilities Safety Board, Washington, D.C.
- Cederdahl, B., and D. Freeman, 1994, internal memorandum, "Division Critical Item - RBOF Regeneration System Improvements," Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina, October 28.
- Clark, B. D., 1994, *Vulnerability Assessment Action Plan Closure - RBOF Item 20 (U)*, RRD-RSE-940259, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Conway, J. T., 1995, *Spent Nuclear Fuel at the Savannah River Site (SRS), Trip Report for June 20-21, 1995*, Defense Nuclear Facilities Safety Board, Washington, D.C.
- Conway, J. T., 1996, *Handling of Spent Nuclear Fuel at the Savannah River Site (SRS), Trip Report for August 5-7, 1996*, Defense Nuclear Facilities Safety Board, Washington, D.C.
- Conway, J. T., 1998, "Sludge Milestones," letter to F. Pena (U.S. Department of Energy), Defense Nuclear Facilities Safety Board, Washington, D.C., April 15.

- DNFSB (Defense Nuclear Facilities Safety Board), 1994, *Recommendation 94-1 (DNFSB 94-1), Improved Schedule for Remediation in the Defense Nuclear Facilities Complex*, Washington, D.C.
- DOE (U.S. Department of Energy), 1993, *Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear and other Reactor Irradiated Nuclear Material and Their Environmental, Safety and Health Vulnerabilities*, Volumes I, II and III, Washington, D.C.
- DOE (U.S. Department of Energy), 1994a, *Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities*, Phase I, Washington, D.C.
- DOE (U.S. Department of Energy), 1994b, *Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities*, Phase II, Washington, D.C.
- DOE (U.S. Department of Energy), 1994c, *Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities*, Phase III, Revision B., Washington, D.C.
- DOE (U.S. Department of Energy), 1995a, *Baseline Change Proposal (BCP) for Disassembly Basin Upgrades*, BCP 95-D-158/005, Savannah River Operations Office, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1995b, *Defense Nuclear Facilities Safety Board Recommendation 94-1 Implementation Plan*, Washington, D.C.
- DOE (U.S. Department of Energy), 1995c, *Final Environmental Impact Statement, Interim Management of Nuclear Materials*, DOE/EIS-0220, Savannah River Site, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1995d, *Final Environmental Impact Statement, Interim Management of Nuclear Materials*, DOE/EIS-0220, Savannah River Site, Aiken, South Carolina, October.
- Guy, J. C., 1993, *Storage Solution for Fuel Tubes in the L-Area Vertical Tube Storage (U)*, RRD-RED-93-0217, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Guy, J. C., 1995, *Installation and Testing of New Horizontal Fuel Rack in L-Area Disassembly Basin (U)*, EFR-RSE-950124, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Hochel, R. C., 1995, *Evaluation of Basin Air Radiological Hazard from Fission Gases (U)*, SRT-ADS-95-0198, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Holmes, W., 1995, *L-3.3 Fuel Bundling (U)*, EFR-RSE-950188, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Howell, J. P., 1996, *Corrosion Surveillance in Spent Fuel Storage Pools (U)*, SRT-MTS-96-0080, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- New, I.B., 1994, *Level Monitoring for the Disassembly Basins (U)*, RRD-940155, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- New, I. B., 1995, *SNF Dry Storage Facility Procurement Technical Specifications (U)*, EFR-9500138, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.

- New, I. B., 1996a, *Operational Release of Continuous Deionizer Facilities (U)*, EFR-960058, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- New, I. B., 1996b, *Safety Analysis Report (U)*, EFR-960085, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Potvin, J. C., 1997, WSRC Letter, *DOE Commitment (WSRC Ref.: 97-007) RE: SRS Spent Nuclear Fuel Trip Report Observations- EFR, RBOF, K-Basin, L-Basin (U)*, PEC-DNF-97-0022, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Reed, R. L., 1994, *100 Area Irradiated Fuel Consolidation and Horizontal Storage Criticality Concerns (U)*, SRT-CMA-930074, Revision 0, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Reed, R. L., 1995, *Nuclear Criticality Safety Evaluation: Reactivity Effects of Tilting Fuel Assemblies and Bundles in RBOF (U)*, EPD-CTG-950026, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Reed, R. L., 1996, *Nuclear Criticality Safety Evaluation: Credible Water Depth for Criticality Incidents in RBOF (U)*, N-NCS-H-00011, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Sidey, K., 1997, "Commitment #2, Re: SRS Spent Nuclear Fuel Trip Report Observations- EFR, RBOF, K-Basin, L-Basin," letter to A. Poston, U.S. Department of Energy, Savannah River Operations Office, Aiken, South Carolina, January 16.
- Sindelar, R. L., H. B. Peacock, Jr., P. S. Lam, N. C. Iyer, and M. R. Louthan, Jr., 1996, *Acceptance Criteria for Interim Dry Storage of Aluminum-Alloy Clad Spent Nuclear Fuels (U)*, WSRC-TR-95-0347, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Smith, P. K., 1998, *DOE Vulnerability Assessment Corrective Action Plan Status (U)*, SFS-RSE-98-0081, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.
- Voss, T. J., 1997a, "SRS Spent Nuclear Fuel Trip Report Observations-SFS, RBOF, K-Basin, L-Basin (U)," PEC-DNF-97-0062, letter to A. Poston, U.S. Department of Energy, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina, April 17.
- Voss, T. J., 1997b, "DOE Commitment - SIMS 96-024, Commitment 001 (WSRC Ref.: 97-015) Re: SRS Spent Nuclear Fuel Trip Report Observations- EFR, RBOF, K-Basin, L-Basin (U)," PEC-DNF-97-0031, letter, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina, February 4.
- WSRC (Westinghouse Savannah River Company), 1995a, *Disassembled Component Log - Fuel Bundling Station*, DPSOL 105-3800B-PLK, Savannah River Site, Aiken, South Carolina.
- WSRC (Westinghouse Savannah River Company), 1995b, *Disassembly Basins Upgrades*, M-FPR-G-00003, Revision 3, Savannah River Site, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1995c, *Basis for Interim Operation (BIO) for K-Reactor in Cold Standby (U)*, WSRC-TR-94-0207, Revision 0, Savannah River Site, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1995d, *Hazards Assessment Document for K-Reactor in Cold Standby (U)*, WSRC-TR-93-665, Savannah River Site, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1995e, *Install New Twin Hoist Yoke*, Work Request Package No. BHBR5, Savannah River Site, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1996, *Basis for Interim Operation for the L-Reactor Facility (U)*, WSRC-TR-95-0054, Savannah River Site, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1997a, *SRS Seismic Response Analysis Design Basis Guidelines*, WSRC-TR-97-0085, Revision 0, Savannah River Site, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1997b, *Spent Fuel Storage Division Facility Assessment Checklist, Basin Operations and Fuel Shipment*, Revision 3, Savannah River Site, Aiken, South Carolina.

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APPENDIX C

SPENT NUCLEAR FUEL BACKGROUND AND INVENTORY

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APPENDIX C. SPENT NUCLEAR FUEL BACKGROUND AND INVENTORY

C.1 Background

C.1.1 GENERAL CHARACTERISTICS

In nuclear reactors a process occurs known as the fission chain reaction, in which atomic nuclei in reactor fuel respond to collisions with neutrons by splitting into two major fragments and two or three neutrons. The neutrons can interact with other fuel nuclei, thereby continuing the chain reaction.

In comparison to a chemical reaction involving the same mass, a nuclear reaction releases a large amount of energy, mostly the kinetic energy of the fission fragments and neutrons and the subsequent radioactive decay of the fission fragments (fission products). This energy makes nuclear fission an attractive source of energy for commercial power producers. DOE operated its production reactors principally because the neutrons caused nonfission nuclear reactions of interest to national defense (i.e., isotopic transmutation). Research reactors use the fission process to produce medical isotopes or for other research purposes.

Nuclear fuel must contain atoms that can be fissioned (called fission atoms). Fission atoms are fissioned by low-energy (thermal) neutrons. Therefore, to maintain the chain reaction, the high energy, fast neutrons produced by fission must be slowed to low-energy, thermal neutrons. The process for slowing down the neutrons is called moderation: water, graphite, and heavy water are used as moderators.

Uranium-235 is the fissile atom used most often for nuclear fuel; however, other fissile materials (uranium-233, plutonium-239, and plutonium-241) can be used in nuclear reactors. Uranium-235 represents only about 0.7 percent of the atoms of natural uranium, which is primarily uranium-238. Therefore, many reactors use fuel that has an enriched uranium-235 content. Commercial power reactors typically use fuels

enriched to approximately 2 to 4 percent. Non-commercial reactors, depending on their purpose, use fuel enriched to as much as 93 percent uranium-235. Low-enriched uranium (LEU) has an enrichment below 20 percent; highly enriched uranium (HEU) is enriched 20 percent or higher. The fuels discussed in this EIS are primarily highly enriched uranium fuels.

The uranium in nuclear fuels generally is clad with a metal to protect it from chemical reactions with the moderator water and to prevent the release of fission products to the water. Zirconium, stainless steel, and aluminum are common cladding materials. Most of the SNF analyzed in this EIS (about 48 metric tons heavy metal [MTHM]) is aluminum-clad; the remainder is clad with stainless steel or zirconium.

Inside the cladding, the fuel is often in the form of a ceramic, an alloy that combines uranium with aluminum, metallic uranium, or a uranium oxide or silicide. The fuel can be assembled as parallel plates, concentric tubes, bundles of rods or pins, or other designs. Each assembly has mounting and lifting hardware, structures to direct coolant and moderator flow, and in some cases the capability to install neutron absorbing material and instrumentation. Usually a number of fuel assemblies make up a complete reactor core.

Spent nuclear fuel (SNF) is fuel that has been irradiated in a reactor and contains fissile atoms and fission products. SNF management must consider four fuel characteristics: radiation fields, heat generation, criticality, and chemical stability (corrosion resistance). As the fuel is irradiated in a reactor, much of the uranium is burned, resulting in the production of fission products. These fission products are radioactive; that is, they do not undergo fission but they radiate energy and transmute to other elements. SNF has very high radiation fields, especially for a period of time immediately after it is removed from the reactor. After a period of de-

cay, as the short-lived fission products decay away, the radiation fields decrease, but the fuel is still highly radioactive and requires management for many years.

The heat from the radioactive decay of fission products (decay heat) can produce very high temperatures, requiring fuel recently removed from a reactor to be placed in underwater storage for cooling. Without active cooling, the fuel could overheat and melt or damage the cladding. After a sufficient cooling time that depends on the burnup of the fuel and its composition, fuel assemblies can be stored dry. Dry fuel storage technologies must be designed to release residual decay heat.

Long-term storage of SNF in water can lead to corrosion of the fuel cladding. Careful control of water chemistry can reduce the rate of corrosion. Aluminum-clad fuels, which are considered in this EIS, are more prone to corrosion in water than are stainless-steel or zirconium-clad fuels.

Most SNF could undergo a fission chain reaction. However, the fuel density, geometry, temperature, and moderation must support fission, or the chain reaction would not occur because too many neutrons would be absorbed or otherwise lost.

When a reactor is producing enough neutrons to support a chain reaction, it is termed "critical." Criticality occurs when fissile material begins to undergo a chain reaction. SNF management must consider the potential of the fuel to create an unwanted criticality.

SNF can be chemically processed to recover transmitted isotopes for defense or commercial purposes and the fissile and fertile material for conversion into more nuclear fuel.

C.1.2 RECENT SPENT FUEL MANAGEMENT ACTIONS

In 1992, DOE decided to phase out defense-related SNF processing. Subsequently, the Department began to establish programs to manage DOE SNF that were no longer based on the production of strategic nuclear material. DOE identified the initial components of this plan in the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995a) (hereafter referred to as the Programmatic SNF EIS). The Record of Decision for this environmental impact statement (EIS) (60 FR 28680) stated in part that DOE would consolidate the management of its aluminum-clad SNF at the Savannah River Site (SRS) and would consolidate nonaluminum-clad fuels at the Idaho National Engineering and Environmental Laboratory. As a result, about 20 MTHM of stainless-steel and zirconium-clad SNF stored at SRS was designated for shipment to the Idaho National Engineering and Environmental Laboratory for management. In addition, DOE decided to ship about 10 MTHM of aluminum-clad SNF to SRS from domestic, and DOE research reactors, and the Idaho National Engineering and Environmental Laboratory.

However, in the Programmatic SNF EIS Record of Decision DOE made no decisions on the technologies it would apply to the management of SNF at the designated storage sites. The Record of Decision stated that the selection of SNF stabilization technologies and the preparation of SNF for ultimate disposition would be the subject of site-specific and fuel-type-specific evaluations prepared in accordance with the National Environmental Policy Act and tiered from the Programmatic SNF EIS (DOE 1995a).

In October 1995, DOE assessed the environmental impacts of stabilizing certain nuclear materials at SRS that presented potential environment, safety, and health vulnerabilities (DOE 1995b). The material evaluated by DOE included SRS production reactor SNF stored in the reactor disassembly basins and research re-

actor SNF stored in the Receiving Basin for Offsite Fuel. The Department decided to stabilize SNF that presented potential environmental, safety, and health vulnerabilities by processing the material through the existing chemical separations facilities at SRS. Under these decisions (60 FR 65300, 61 FR 6633, and 62 FR 17790), about 175 MTHM of the approximately 195 MTHM of SNF at SRS will be stabilized. After stabilization, the resulting material will be treated and managed so that it is acceptable for permanent disposition once those decisions are made. DOE concluded the remaining material, all of which was stored in the Receiving Basin for Offsite Fuel, was stable and could remain as is for several years pending disposition decisions. In addition, DOE decided some of the stable material might have programmatic value, that is, be of use to future DOE missions. Mark-18 targets stored in the Receiving Basin for Offsite Fuel could be shipped to other DOE sites for programmatic uses, including irradiation for transuranium isotope production (primarily for National Aeronautics and Space Administration use) and defense stockpile stewardship activities.

In May 1996 DOE issued its Record of Decision (61 FR 25092) for the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor [FRR] Spent Nuclear Fuel* (DOE 1996a) (hereafter referred to as the FRR EIS). The Department decided to accept and manage foreign research reactor SNF that contains uranium enriched in the United States. In keeping with its 1995 programmatic decision (60 FR 28680), DOE decided it would manage the aluminum-clad portion of the foreign research reactor SNF, about 18 MTHM, at SRS. Under the foreign research reactor receipt program, shipments from foreign reactors to SRS began in September 1996 and could continue for as long as 13 years. At present, SRS is receiving this fuel in either the Receiving Basin for Offsite Fuel or the L-Reactor Disassembly Basin. Figure C-1 shows projected receipts of aluminum-clad SNF at SRS from foreign and domestic sources, based on 1996 estimates. Because some countries may choose not to partici-

pate in the return of foreign SNF, the amount of aluminum-based foreign SNF to be managed at SRS may be less.

The May 1996 decision to accept foreign research reactor SNF for management in the United States (61 FR 25092) stated that DOE would issue a separate Record of Decision, after appropriate environmental reviews, to announce its plans for the management of such fuel. The Department committed to the aggressive pursuit of one or more new packaging or non-processing technologies that would put foreign research reactor SNF in a form or container suitable for disposal in a geologic repository. DOE also committed to place foreign research reactor SNF in dry storage at SRS (after required treatment or packaging) pending offsite storage or disposal. DOE also stated that if a new treatment technology was not ready for implementation by 2000, DOE would consider the chemical separation of some foreign reactor SNF that would blend the material down to low-enriched uranium in F Canyon at SRS. DOE might then place it under International Atomic Energy Agency safeguards.

C.2 Inventory

C.2.1 PHYSICAL INVENTORY

As Figure C-1 indicates, most SNF receipts would occur before 2015; however, SRS will continue to receive small amounts over the entire period of analysis (until 2035). There is great variety in the SNF that SRS must manage over the next 40 years. Therefore, DOE has categorized the SNF into six groups to facilitate analysis. The SNF in each category should receive nearly identical management. The basis for the categorization was often the size of the fuel in relation to packaging; however, other considerations were included such as physical characteristics, chemical characteristics, and radionuclide content. For example, one category includes all SNF in powder form. The following subsections describe the six fuel groups and list the SNF inventory associated with each group. Receipts per year will be approximately 150 Materials Test

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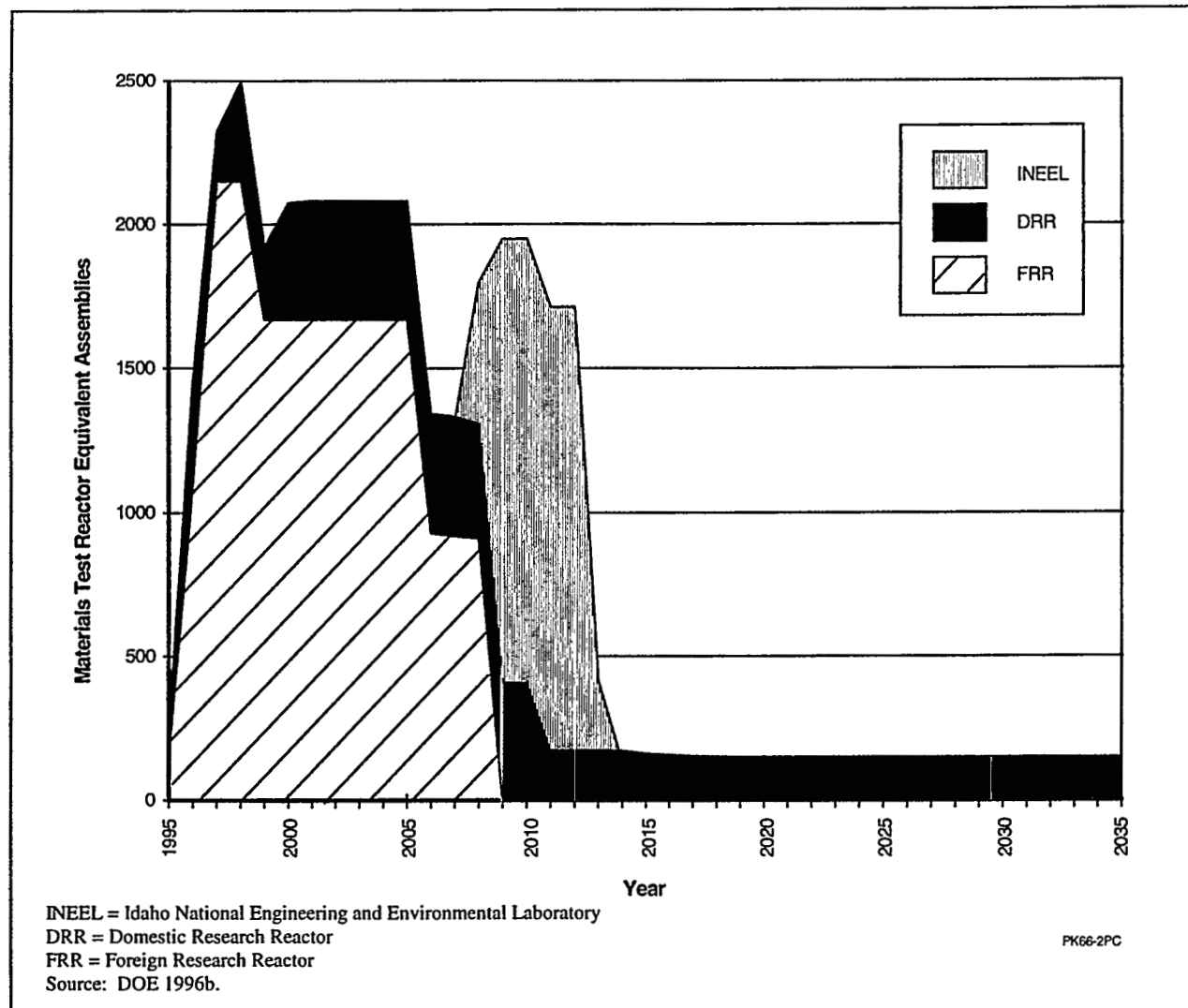


Figure C-1. Projected receipts of SNF at the Savannah River Site.

Reactor-like Elements from domestic reactors and 12 High Flux Isotope Reactor assemblies from Oak Ridge.

C.2.1.1 Group A: Uranium and Thorium Metal Fuels

Group A consists mostly of chemically reactive uranium and thorium metal fuels. Many of the fuel elements are declad, and much of this group consists of depleted or natural uranium. As indicated in Table C-1, Group A fuels consist of four fuel types. The Experimental Breeder Reactor-II Blanket Fuels have been declad and the depleted uranium slugs placed in aluminum cans. The Advanced Reactivity Measurement

Facility (ARMF) Core Filter Block is a 6 × 6 × 24-inch (15.2 × 15.2 × 61-centimeter) block of depleted uranium. The Sodium Reactor Experiment fuel consists of declad thorium metal placed in 3.5-inch (8.9-centimeter) diameter by 110-inch (279-centimeter) long cans. The Mark-42 targets are unirradiated tubes of plutonium oxide in an aluminum matrix approximately 3.7 inches (9.4 centimeters) in diameter and 168 inches (426 centimeters) long.

C.2.1.2 Group B: Materials Test Reactor-Like Fuels

Group B is comprised mostly of Materials Test Reactor fuels, as described in Section 1.5 and

Figure 1-3, plus a few other fuels of similar size and composition. Table C-2 lists the Group B inventory

Table C-1. Inventory of Group A SNF.

Name	Items	Units	Location
Experimental Breeder Reactor Blankets	59	Cans	SRS Wet Basins ^a
ARMF Core Filter Block	1	Filter	INEEL
Sodium Reactor Experiment	36	Cans	SRS Wet Basins
Mark-42	16	Bundles	SRS Wet Basins

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.
ARMF = Advanced Reactivity Measurement Facility
INEEL = Idaho National Engineering and Environmental Laboratory

C.2.1.3 Group C: HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging

Group C fuels are similar in composition to Group B fuels in that they are aluminum-clad, highly enriched uranium and low enriched uranium oxides and silicides, but their size or shape precludes packaging without resizing or special packaging considerations. Some of the Group C SNF is smaller in diameter and longer than Group B fuels. Other fuel in this group is larger than Group B fuels in both diameter and length and often comes in odd shapes such as 0.5-by-0.9-meter (1.5-by-3-foot) cylinders or spheres with a diameter of 74 centimeters (29 inches). Table C-3 lists Group C inventory.

C.2.1.4 Group D: Loose Uranium Oxide in Cans

Group D fuels consist of loose uranium oxide and fission products in aluminum cans. Table C-4 lists the Group D inventory.

The Sterling Forest Oxide material in this fuel group is a residue of highly enriched uranium, plutonium, fission products, mixed oxides (chromium, nickel, iron, barium), barium acetate, and barium nitrate which resulted from the production of medical isotopes (primarily molybdenum 99). The material was plated on the inside of stainless steel tubes when it was irradiated. The material was then removed from the tubes with an acid flush and the uranium was recovered from a nitrate-sulfate solution, after eliminating the sulfate by precipitating with barium acetate and filtering. The filtrate was evaporated and pyrolyzed at 300°C to an oxide form in an aluminum can. The can was sealed and shipped to SRS where it was placed into storage in the Receiving Basin for Offsite Fuels. Both the can and the oxide powder it contains are readily dissolved in acid.

The other items in this fuel group are liquid targets that DOE assumes would be converted to oxide prior to shipment to SRS.

C.2.1.5 Group E: Higher Actinide Targets

Group E contains target materials used to generate radionuclides with atomic numbers beyond that of uranium. The targets are placed in nuclear reactors and irradiated with neutrons, which causes nonfission nuclear reactions. These targets are aluminum-clad plutonium oxide that contain significant quantities of americium and curium, which react under neutron irradiation to produce even higher atomic number elements such as californium. Table C-5 lists the Group E inventory.

C.2.1.6 Group F: Non-aluminum Clad Fuels

Group F comprises the large variety of non-aluminum-clad SNF at SRS that DOE must ship to the Idaho National Engineering and Environmental Laboratory under the Record of Decision for the Programmatic SNF EIS (DOE 1995a). Table C-6 lists the Group F inventory.

C.2.2 RADIONUCLIDE INVENTORY

The six SNF groups that DOE would manage at SRS possess diverse chemical, physical, and radiological characteristics. There is also diversity within any single fuel group. In the absence of detailed radionuclide characterization of the fuel, DOE has simplified the analyses for this EIS by developing an analytical construction called a Reference Fuel Assembly (Garrett et al. 1995). The Reference Fuel Assembly is used as a standard reference for scaling fuel group characteristics. This assembly is a composite of depleted uranium, highly enriched uranium, and special target radionuclides.

To determine the radionuclide inventories of each fuel group, DOE calculated the ratio of radioactivity of each nuclide in the Reference

Fuel Assembly to the fissile mass of the Reference Fuel Assembly; multiplied the resulting ratios by the fissile mass of the fuel groups to obtain nuclide-specific inventories for each fuel group. DOE performed an identical calculation based on total heavy metal mass rather than fissile mass of each fuel group. DOE conservatively used the calculation (fissile mass ratio or total heavy metal mass ratio) that yielded the largest value of each radionuclide to calculate the inventory of each radionuclide for the fuel group. Scaling by fissile mass is important because the fission products potentially produce most of the radiological impacts. Scaling by heavy metal mass is important because heavy metal mass is an indicator of processing time and provides appropriate representation of Group A fuels which contain little fissile mass. Table C-7 lists the results of these calculations.

Table C-2. Inventory of Group B SNF.

Name	Items	Units	Location
FRR MTR	10,812	Assemblies	Argentina, Australia, Austria, Brazil, Canada, Chile, Colombia, France, Germany, Greece, Indonesia, Iran, Israel, Italy, Japan, Netherlands, Pakistan, Peru, Philippines, Portugal, South Africa, Spain, Sweden, Switzerland, Taiwan, Thailand, Turkey, United Kingdom, Uruguay, Venezuela
DRR MTR	11,799	Assemblies	LANL, INEEL, ORNL, Brookhaven National Laboratory, Massachusetts Institute of Technology, Georgia Institute of Technology, Iowa State University, University of Massachusetts-Lowell, University of Michigan, Missouri University, Purdue University, Rhode Island Nuclear Center, University of Virginia, Worcester, National Institute of Standards and Technology
MTR	Approximately 1,100	Assemblies	SRS Wet Basins ^a
Cylindrical MTR	145	Assemblies	Japan
Box MTR	28	Assemblies	Japan
Tube MTR	4,077	Assemblies	Australia, Belgium, Denmark, Germany
Missouri University Research Reactor	224	Assemblies	SRS Wet Basins, MURR, INEEL
Advanced Test Reactor	3,132	Assemblies	INEEL
Advanced Reactivity Measurement Facility	67	Assemblies	INEEL
University of Washington	26	Bundles	INEEL
Advanced Reactivity Measurement Facility Plate	15	Plates	INEEL
Sterling Forest Fuel	200	Assemblies	SRS Wet Basins

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

b. This value changes with FRR and DRR ongoing receipts. Some double counting with FRR and DRR entries exists.

LANL = Los Alamos National Laboratory

ORNL = Oak Ridge National Laboratory

MTR = Materials Test Reactor

FRR = Foreign Research Reactor

MURR = Missouri University Research Reactor

DRR = Domestic Research Reactor

INEEL = Idaho National Engineering and Environmental Laboratory

Table C-3. Inventory of Group C SNF.

Name	Items	Units	Location
Mark-14	1	Can	SRS Wet Basins ^a
Oak Ridge Research Reactor	165	Assemblies	SRS Wet Basins
HWCTR	1	Can	SRS Wet Basins
Pin bundle	12	Bundles	Canada, Jamaica
Pin cluster	2,792	Clusters	Canada, South Korea
ZPTR	45	Assemblies	Cornell University
ZPR	17	Assemblies	Manhattan University
OSR	24	Assemblies	Ohio State
Argonaut	50	Assemblies	Florida
Reactor a-Haut Flux	90	Assemblies	SRS Wet Basins, France
High Flux Isotope Reactor	540	Assemblies	ORNL
High Flux Isotope Reactor	1	Can	SRS Wet Basins
BSR	32	Assemblies	ORNL/SRS
Tower Shielding Reactor	1	Element	ORNL
Tower Shielding Reactor	2	Cans	ORNL
Sandia Pulse Reactor	43	Assemblies	Sandia National Laboratories
Oak Ridge Reactor	9	Cans	ORNL/SRS

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

BSR = Bulk Shielding Reactor

OSR = Ohio State Reactor

ZPR = Zero Power Reactor

ZPTR = Zero Power Test Reactor

ORNL = Oak Ridge National Laboratory

HWCTR = Heavy Water Components Test Reactor

Table C-4. Inventory of Group D SNF.

Name	Items	Units	Location
Sterling Forest Oxide	676	Cans	SRS Wet Basins ^a
Other non-MTR targets	6,750	Cans	Canada, Belgium, Argentina, Indonesia

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

MTR = Materials Test Reactor

Table C-5. Inventory of Group E SNF.

Name	Items	Units	Location
Mark-18	65	Assemblies	SRS Wet Basins ^a
Mark-51	60	Slugs	SRS Wet Basins
Other	114	Slugs	SRS Wet Basins

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

Table C-6. Inventory of Group F SNF.

Name	Items	Units	Current Location
Carolinas-Virginia Tube Reactor	3	Bundles	SRS Wet Basins ^a
Dresden	24	Sleeves	SRS Wet Basins
Dresden	6	Cans	SRS Wet Basins
Elk River Reactor	38	Bundles	SRS Wet Basins
LWR Samples	5	Cans	SRS Wet Basins
H. B. Robinson	1	Can	SRS Wet Basins
Saxton	13	Bundles	SRS Wet Basins
Saxton	3	Cans	SRS Wet Basins
Saxton	3	Test Tubes	SRS Wet Basins
Vallecitos	2	Bundles	SRS Wet Basins
Babcock & Wilcox Scrap	1	Can	SRS Wet Basins
EBR-II (ANL-MXOX)	1	Cans	SRS Wet Basins
EBWR	6	Cans	SRS Wet Basins
EBWR	4	Bundles	SRS Wet Basins
EBWR	288	Assemblies	SRS Wet Basins
EPRI	1	Can	SRS Wet Basins
GCRE	6	Cans	SRS Wet Basins
GCRE	66	Assemblies	SRS Wet Basins
HWCTR	34	Slugs	SRS Wet Basins
HWCTR	87	Cans	SRS Wet Basins
HWCTR	57	Assemblies	SRS Wet Basins
HWCTR	22	Bundles	SRS Wet Basins
HWCTR	9	Test Tubes	SRS Wet Basins
HTRE	13	Cans	SRS Wet Basins
ML-1	68	Assemblies	SRS Wet Basins
ORNL S1W-1 rods	3	Cans	SRS Wet Basins
ORNL Mixed Oxide (BW-1)	1	Can	SRS Wet Basins
Shippingport	127	Pins	SRS Wet Basins
SPERT-3	3	Cans	SRS Wet Basins
Sodium Reactor Experiment (Carbide)	1	Can	SRS Wet Basins
CANDU	3	Cans	SRS Wet Basins
CANDU	56	Rods	SRS Wet Basins

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

EBR = Experimental Breeder Reactor

HWCTR = Heavy Water Components Test Reactor

CANDU = Canadian Deuterium-Uranium Reactor

LWR = Light Water Reactor

EBWR = Experimental Boiling Water Reactor

ANL-MXOX = Argonne National Laboratory Mixed Oxide

ORNL = Oak Ridge National Laboratory

GCRE = Gas Cooled Reactor Experiment

HTRE = High Temperature Reactor Experiment

ML-1 = Mobile Low Power Plant No. 1

SPERT-3 = Special Power Excursion Test-3

EPRI = Electric Power Research Institute

*Spent Nuclear Fuel Background and Inventory***Table C-7. Radionuclide inventories based on the Reference Fuel Assembly (curies).**

Nuclide ^a	Reference Fuel	Fuel Group					
	Assembly	A	B	C	D	E	F
H-3	51.6	2540	144,000	46,000	9,090	112	9,780
Kr-85	1,050	51,700	2,920,000	935,000	185,000	2,270	199,000
Sr-89	49.2	2420	137,000	43,800	8,670	107	9,320
Sr-90	8,080	398,000	22,500,000	7,200,000	1,420,000	17,500	1,530,000
Y-90	8,080	398,000	22,500,000	7,200,000	1,420,000	17,500	1,530,000
Y-91	213	10,500	593,000	190,000	37,500	461	40,400
Zr-95	454	22,400	1,260,000	404,000	80,000	983	86,000
Nb-95	1,010	49,800	2,810,000	899,000	178,000	2,190	191,000
Nb-95m	3.37	166	9,390	3,000	594	7.30	639
Tc-99	1.03	50.7	2,870	917	181	2.23	195
Rh-103m	1.96	96.6	5,460	1,750	345	4.25	371
Rh-106	21,100	1,040,000	58,800,000	18,800,000	3,720,000	45,700	4,000,000
Ru-103	2.17	107	6,040	1,930	382	4.70	411
Ru-106	21,100	1,040,000	58,800,000	18,800,000	3,720,000	45,700	4,000,000
Ag-110	2.32	114	6,460	2,070	409	5.03	440
Ag-110m	174	8,570	485,000	155,000	30,700	377	33,000
Cd-113m	6.95	342	19,400	6,190	1,220	15.1	1,320
Sn-119m	3.93	194	10,900	3,500	693	8.51	745
Sn-123	14.5	714	40,400	12,900	2,560	31.4	2,750
Sb-125	870	42,900	2,420,000	775,000	153,000	1,880	165,000
Te-125m	212	10,400	590,000	189,000	37,400	459	40,200
Te-127	34.7	1,710	96,600	30,900	6,110	75.2	6,570
Te-127m	35.4	1,740	98,600	31,500	6,240	76.7	6,710
Te-129	0.0012	591	3.34	1.07	0.211	0.0026	0.227
Te-129m	0.00185	911	5.15	1.65	0.326	0.00401	0.351
Cs-134	10,300	507,000	28,700,000	9,170,000	1,810,000	22,300	1,950,000
Cs-137	9,280	457,000	25,800,000	8,260,000	1,640,000	20,100	1,760,000
Ba-137m	8,780	433,000	24,500,000	7,820,000	1,550,000	19,000	1,660,000
Ce-141	0.0646	3.18	180	57.5	11.4	0.140	12.2
Ce-144	47,800	2,350,000	133,000,000	42,600,000	8,420,000	104,000	9,060,000
Pr-144	47,800	2,350,000	133,000,000	42,600,000	8,420,000	104,000	9,060,000
Pr-144m	574	28,300	1,600,000	511,000	101,000	1,240	109,000
Pm-147	18,800	926,000	52,400,000	16,700,000	3,310,000	40,700	3,560,000
Pm-148m	0.00893	0.44	24.9	7.95	1.57	0.0193	1.69
Sm-151	69.4	3,420	193,000	61,800	12,200	150	13,100
Eu-154	727	35,800	2,020,000	647,000	128,000	1,570	138,000
Eu-155	381	18,800	1,060,000	339,000	67,100	825	72,200
Tl-208	8.46	417	23,600	7,530	1,490	18.3	1,600
Pb-209	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Pb-211	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Pb-212	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Bi-211	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Bi-212	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Bi-213	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Po-212	15.1	744	42,100	13,400	2,660	32.7	2,860
Po-213	0.00855	0.421	23.8	7.61	1.51	0.0185	1.62

Table C-7. (continued).

Nuclide ^a	Reference Fuel	Fuel Group					
	Assembly	A	B	C	D	E	F
Po-215	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Po-216	23.6	1,160	65,700	21,000	4,160	51.1	4,470
At-217	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Rn-219	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Rn-220	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Fr-221	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Ra-223	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Ra-224	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Ra-225	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Ac-225	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Ac-227	0.0171	0.842	47.6	15.2	3.01	0.037	3.24
Th-227	0.0164	0.808	45.7	14.6	2.89	0.0355	3.11
Th-228	23.5	1160	65,500	20,900	4,140	50.9	4,450
Th-229	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Th-231	0.0114	0.562	31.8	10.2	2.01	0.0247	2.16
Th-232	0.0172	0.847	47.9	15.3	3.03	0.0373	3.26
Th-234	0.000216	0.0106	0.60	0.192	0.0381	0.000468	0.0405
Pa-231	0.228	11.2	635	203	40.2	0.494	43.2
Pa-233	0.0859	4.23	239	76.5	15.1	0.186	16.3
Pa-234m	0.000216	0.0106	0.60	0.192	0.0381	0.000468	0.0405
U-232	40.9	2,010	114,000	36,400	7,210	88.6	7,750
U-233	39.3	1,940	109,000	35,000	6,930	85.1	7,450
U-234	1.92	94.6	5,350	1,710	338	4.16	364
U-235	0.0114	0.562	31.8	10.2	2.01	0.0247	2.16
U-236	0.0329	1.62	91.6	29.3	5.80	0.0713	6.23
U-237	0.259	12.8	721	231	45.6	0.561	49.1
U-238	0.0842	4.15	235	75.0	14.8	0.182	16.0
Np-237	0.00881	0.434	24.5	7.85	1.55	0.02	1.67
Np-239	9.62	474	26,800	8,570	1,700	21	1,820
Pu-236	112	5,520	312,000	99,700	19,700	250	21,200
Pu-238	51.9	2,560	145,000	46,200	9,150	340	9,830
Pu-239	58	2,860	162,000	51,700	10,200	130	11,000
Pu-240	9,780	482,000	27,200,000	8,710,000	1,720,000	23,000	1,850,000
Pu-241	10,600	522,000	29,500,000	9,440,000	1,870,000	23,000	2,010,000
Am-241	51.7	2,550	144,000	46,000	9,110	450	9,800
Am-242	0.34	16.7	947	303	59.9	0.74	64.4
Am-242m	0.341	16.8	950	304	60.1	0.74	64.6
Am-243	9.62	474	26,800	8,570	1,700	21	1,820
Cm-242	490	24,100	1,360,000	436,000	86,300	1,100	92,800
Cm-243	4.9	241	13,600	4,360	863	11	928
Cm-244	2,750	135,000	7,660,000	2,450,000	485,000	18,000	521,000
Cm-246	0.215	10.6	599	191	37.9	150	40.7
Totals	231,000	11,400,000	644,000,000	206,000,000	40,700,000	520,000	43,800,000

a. Refer to Table C-8 for the names of the elements.

Table C-8. Chemical symbols used in Table C-7 and the corresponding element names.

H-3	=	tritium
Kr	=	krypton
Sr	=	strontium
Y	=	yttrium
Zr	=	zirconium
Nb	=	niobium
Tc	=	technetium
Rh	=	rhodium
Ru	=	ruthenium
Ag	=	silver
Cd	=	cadmium
Sn	=	tin
Sb	=	antimony
Te	=	tellurium
Cs	=	cesium
Ba	=	barium
Ce	=	cerium
Pr	=	praseodymium
Pm	=	promethium
Sm	=	samarium
Eu	=	europium
Tl	=	thallium
Pb	=	lead
Bi	=	bismuth
Po	=	polonium
At	=	astatine
Rn	=	radon
Fr	=	francium
Ra	=	radium
Ac	=	actinium
Th	=	thorium
Pa	=	protactinium
U	=	uranium
Np	=	neptunium
Pu	=	plutonium
Am	=	americium
Cm	=	curium

References

- DOE (U.S. Department of Energy), 1995a, *Final Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement*, DOE/EIS-0203, Idaho Operations Office, Idaho Falls, Idaho.
- DOE (U.S. Department of Energy), 1995b, *Final Environmental Impact Statement, Interim Management of Nuclear Materials*, DOE/EIS-0220, Savannah River Site, Aiken, South Carolina.
- DOE (U.S. Department of Energy), 1996a, *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Research Reactor Spent Nuclear Fuel*, DOE/EIS-0218F, Washington, D.C.
- DOE (U.S. Department of Energy), 1996b, *Technical Strategy for the Treatment, Packaging, and Disposal of Aluminum Based Spent Nuclear Fuel, A Report of the Research Reactor Spent Nuclear Fuel Task Team*, Volume I, Washington, D.C.
- Garrett, R. L., S. D. Kahook, L. R. Canas, and M. J. Beckum, 1995, "Spent Nuclear Fuel Characterization for a Bounding Reference Assembly for the Receiving Basin for Offsite Fuel," *Nuclear Safety*, 36:2, U.S. Department of Energy Office of Scientific and Technical Information, Oak Ridge, Tennessee, July-December.

APPENDIX D. ACCIDENT ANALYSIS

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APPENDIX D. ACCIDENT ANALYSIS

This appendix provides detailed information on the management of spent nuclear fuel (SNF). The information includes potential accident scenarios for new, modified, and existing facilities that the U.S. Department of Energy (DOE) would use for each alternative. The appendix provides estimates of the quantity and composition of hazardous materials that could be released in an accident as well as the consequences to workers and the public, estimated in terms of dose and latent cancer fatalities for radiological releases and of concentration levels for chemical releases.

The sources of information for the accident analyses for existing facilities are safety analysis reports and basis for interim operation documents. For new or modified facilities the sources vary, depending on the processes involved. In general, DOE performed detailed hazard assessments to identify potential significant accidents, basing the determination of significance on the existence of sufficient energy sources and hazardous materials that, if released, would impact workers or the public. The following sections provide specific information on the hazards assessments for the alternatives.

D.1 General Accident Information

An accident, as discussed in this appendix, is an inadvertent release of radiological or chemical hazardous materials as a result of a sequence of one or more probable events. The sequence usually begins with an initiating event, such as a human error and explosion, or earthquake and structural failure, followed by a succession of other events that could be dependent or independent of the initial event, which dictate the accident's progression and the extent of materials released. Initiating events fall into three categories:

- *Internal initiators* – normally originate in and around the facility but are always a result of facility operations. Examples in-

clude equipment or structural failures, human errors, and internal flooding.

- *External initiators* – are independent of facility operations and normally originate from outside the facility. Some external initiators affect the ability of the facility to maintain its confinement of hazardous materials because of potential structural damage. Examples include aircraft crashes, nearby explosions, and toxic chemical releases at nearby facilities that affect worker performance.
- *Natural phenomena initiators* – are natural occurrences that are independent of facility operations and occurrences at nearby facilities or operations. Examples include earthquakes, high winds, floods, lightning, and snow. Although natural phenomena initiators are independent of external facilities, their occurrence can involve those facilities and compound the progression of the accident.

The likelihood of an accident occurring and its consequences usually depend on the initiator and the sequence of events and their frequencies or probabilities. Accidents can be grouped into four categories—anticipated, unlikely, extremely unlikely, and not reasonably foreseeable, as listed in Table D-1.

DOE based the frequencies of accidents at existing SNF management facilities on safety analyses and historical data about event occurrences. For proposed new facilities without design details, DOE based the accident frequencies on hazard analyses, historical data for similar facilities and operations, and best estimates. For all facilities, DOE analyzed the bounding accident in appropriate accident classes (e.g., natural phenomena hazards, operational errors, external events), such as the worst case fire, to represent all other accident in that class.

Table D-1. Accident frequency categories.

Accident Frequency category	Frequency range (occurrences per year)	Description
Anticipated	Less than once in 10 years but greater than once in 100 years	Accidents that might occur several times during facility lifetime
Unlikely	Less than once in 100 years but greater than once in 10,000 years	Accidents that are not likely to occur dur- ing facility lifetime; natural phenomena include Uniform Building Code-level earthquake, maximum wind gust, etc.
Extremely unlikely	Less than once in 10,000 years but greater than once in 1,000,000 years	Accidents that probably will not occur during facility life cycle; this includes the design-basis accidents
Beyond extremely unlikely	Less than once in 1,000,000 years	All other accidents

TC

EC
TC

Source: DOE (1994).

D.2 Accident Analysis Method

DOE tailored the methods it used to analyze potential accidents to the specific situation. For accidents that could result from operations at existing facilities, the analysis used the applicable impacts described in existing safety analysis documents. For these facilities the analysis included no new modeling; through a screening process, it included only accident scenarios pertaining to operations related to SNF management. Depending on the alternative, one or more new facilities or major modifications to existing facilities could be required. For example, a new Transfer and Storage Facility would be common to many, but not all, of the alternatives. Some alternatives would require the construction of a new treatment component to operate in conjunction with the Transfer and Storage Facility. For these new facilities, hazard analyses were performed to identify bounding accident scenarios, as explained below. The identified accidents were modeled for radiological impacts (Simpkins 1997) using the AXAIRQ computer code (Simpkins 1995a,b), which is described in this section.

The accident sequences analyzed in this EIS would occur at frequencies generally greater

than once in 1,000,000 years. However, the analysis considered accident sequences with smaller frequencies if their impacts could provide information important to decisionmaking.

D.2.1 TECHNOLOGIES AND RELATED FACILITIES

DOE has organized the accident data in this appendix by the facilities it would use for each alternative. Table D-2 lists the technologies and the corresponding facilities that DOE would use to implement each. DOE organized the accident impacts in Chapter 4 by technology to reflect potential accident occurrences for the associated facilities listed in Table D-2.

Table D-2 also lists applicable types of fuel that DOE would treat and manage under each alternative. The accident analyses performed for each facility and alternative do not take explicit account of specific fuel properties and characteristics. Rather, the analyses defined a reference fuel assembly (RFA; Appendix C) and furnace batch equivalent (FBE; WSRC 1998) amounts of material at risk (MAR). The FBE MAR was used to analyze all events for all new treatment technologies and the RFA MAR for events related to SRS wet basins and SRS canyons.

TC

Table D-2. Alternatives and corresponding facilities.

Facility	Direct co-disposal	Repackage and prepare to ship	Melt and dilute	Melt and dilute in a renovated reactor	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing	Continued wet storage
F and H Canyons ^a						Yes		Yes	
Receiving Basin for Offsite Fuel ^a	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Reactor Disassembly Basins ^a	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Transfer and Storage Facility	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Electrometallurgical Treatment Facility							Yes		
Melt and Dilute Treatment Facility									
Melt and Dilute in a Renovated Reactor ^a			Yes	Yes					
Mechanical Dilution Treatment Facility					Yes ^b				
Vitrification Facilities:									
Glass Material Oxidation and Dissolution System						Yes ^c			
Plasma Arc						Yes ^c			
Dissolve and Vitrify, New Facility						Yes ^d			
Dissolve and Vitrify, F Canyon						Yes ^d			
Applicable Fuel Type ^e	A, B, C	E, F	A, B, C, D	A, B, C, D	B, C	A, B, C, D	A, B, C, D	A, B, C, D, E	A, B, C, D, E

a. Existing facilities. All others are new or modified facilities.

b. Mechanical dilution is by either chop and dilute or press and dilute.

c. F Canyon is not required for Glass Material Oxidation and Dissolution System (GMODS) and Plasma Arc Treatment processes.

d. F Canyon is required to dissolve and dilute SNF before vitrification.

e. Fuel types: A - uranium and thorium metal fuels, B - Materials Test Reactor-like fuel, C - highly enriched uranium/low enriched uranium oxides and silicides requiring resizing or special packaging, D - loose uranium oxide in cans, E - higher-actinide targets, F - non-aluminum-clad fuels.

D.2.2 RADIOLOGICAL HAZARDS

The analysis used the computer code AXAIRQ to model accidental atmospheric radioactive releases from the Savannah River Site (SRS) that are of relatively short duration. AXAIRQ strictly follows the guidance in Regulatory Guide 1.145 (NRC 1982) on accidental releases, and has been verified and validated. The release can originate from a vent or stack and release heights can be from 0 to 100 meters (0 to 328 feet), as appropriate for the scenario. The code considers terrain for elevated releases. In accordance with the regulatory guide, it considers plume meander and fumigation under certain conditions. Plume rise due to buoyancy or momentum is not available. The program uses a 5-year meteorological data base for the Savannah River Site, and determines the shortest distance to the Site boundary in each of the 16 sectors by determining the distance to one of 875 locations along the boundary. The code uses the shortest distance in each sector to calculate the concentration for that sector. DOE used PRIMUS, which was developed by the Oak Ridge National Laboratory, to consider decay and daughter ingrowth.

The analysis assumes that all tritium released would have the form of tritium oxide and, following International Commission on Radiological Protection methodology, the dose conversion factor for tritium has been increased by 50 percent to account for absorption through the skin. For population dose calculations, age-specific breathing rates are applied, but adult dose conversion factors are used. Radiation doses were calculated to the maximally exposed offsite individual (MEI), to the population within 50 miles of the facility, and to an uninvolved worker assumed to be 640 meters (2,100 feet) downwind of the facility.

After DOE calculated the total radiation dose to the public, it used dose-to-risk conversion factors established by the National Council on Radiation Protection and Measurements (NCRP) to estimate the number of latent cancer fatalities that could result from the calculated exposure. No data indicate that small radiation doses cause

cancer; however, to be conservative the NCRP assumes that any amount of radiation has some risk of inducing cancer. DOE has adopted the NCRP factors of 0.0005 latent cancer fatality for each person-rem of radiation exposure to the general public and 0.0004 latent cancer fatality for each person-rem of radiation exposure to radiation workers (NCRP 1993).

D.2.3 CHEMICAL HAZARDS

For chemically toxic materials, the long-term health consequences of human exposure to hazardous materials are not as well understood as those related to radiation exposure. A determination of potential health effects from exposures to chemically hazardous materials, compared to radiation, is more subjective. Therefore, the consequences from accidents involving hazardous materials are expressed in terms of airborne concentrations at various distances from the accident location, rather than in terms of specific health effects.

To determine the potential health effects to workers and the public that could result from accidents involving hazardous materials, the airborne concentrations of such materials released during an accident at varying distances from the point of release were compared to the Emergency Response Planning Guideline (ERPG) values (AIHA 1991). The American Industrial Hygiene Association established these values, which depend on the chemical substance, for the following general severity levels to ensure that the necessary emergency actions occur to minimize exposures to humans.

- ERPG-1 Values. Exposure to airborne concentrations greater than ERPG-1 values for a period greater than 1 hour results in an unacceptable likelihood that a person would experience mild transient adverse health effects or perception of a clearly defined objectionable odor.
- ERPG-2 Values. Exposures to airborne concentrations greater than ERPG-2 values for a period greater than 1 hour results in an unacceptable likelihood that a person would

experience or develop irreversible or other serious health effects or symptoms that could impair a person's ability to take protective action.

- **ERPG-3 Values.** Exposure to airborne concentrations greater than ERPG-3 values for a period greater than 1 hour results in an unacceptable likelihood that a person would experience or develop life-threatening health effects.

Not all hazardous materials have ERPG values. For chemicals that do not have ERPG values, a comparison was made to the most restrictive available exposure limits established by other guidelines to control worker accidental exposures to hazardous materials. In this document, the ERPG-2 equivalent that is used is the PEL-TWA (Permissible Exposure Limit – Time Weighted Average) from 29 CFR Part 1910.1000, Subpart Z.

D.3 Impacts of Postulated Accidents Involving Radioactive Materials

These sections describe the potential accidents and risks associated with the operation of each facility that may be utilized in the implementation of a technology. The impacts of each technology are shown in Sections D.3.5 to D.3.8. The material at risk in all treatments is the same, only the release fractions change. For these cases, over 95 percent of the calculated doses come from the release of plutonium-240 and curium-244. The only exception to this are criticality accident scenarios when over 99 percent of the dose comes from the release of ruthenium-106.

D.3.1 H-CANYON AND FB-LINE

Tables D-3 and D-4 summarize potential accidents and their impacts for the H-Canyon and FB-Line facilities, respectively (WSRC 1993, 1995; TtNUS 1999b).

D.3.2 RECEIVING BASIN FOR OFFSITE FUEL

Potential accidents and their impacts for the Receiving Basin for Offsite Fuel (RBOF) facility have been documented in a safety analysis report (WSRC 1997a). Table D-5 lists the accidents with the highest risks and consequences.

D.3.3 REACTOR DISASSEMBLY BASIN

Potential accidents and their impacts for the L-Reactor Disassembly Basin have been documented in a basis for interim operation report (WSRC 1997b). Table D-6 summarizes the results.

D.3.4 TRANSFER AND STORAGE FACILITIES

DOE could collocate the transfer and storage facilities either in separate buildings or in a single building. The accident impacts associated with the operation of these facilities apply to both cases and assume the location of the facilities in L-Area.

D.3.4.1 Transfer Facility Accidents

Radioactive Material Leaks From Shipping or Storage Cask (TS-1)

Radioactive materials could leak from shipping or storage casks. In this accident sequence, radioactive material would leak to the surface of the shipping or storage cask and a small amount would become airborne. The principal radionuclides would be cesium-137, cerium-144, ruthenium-106, and strontium-90. The total curies released in this scenario would 1.0×10^{-7} and would result in negligible consequences. This event is postulated to occur once in 10 years of operation. The calculated consequences for this scenario are listed in Table D-7.

Table D-3. H-Canyon radiological accidents and impacts.^a

Accident	Maximum curies released	Accident frequency	Accident consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
Ruthenium volatilization	140	Once in 11 years	0.13	0.013	770	0.39
Fire	0.57	Once in 1,600 years	0.53	0.055	3,300	1.6
Earthquake	860	Once in 5,000 years	1.8	0.246	14,000	7.0
Coil and tube, cooling tower circulated	13	Once in 14,000 years	13	1.3	78,000	39
Transfer error to Building 211-H	3,700	Once in 14,000 years	1.5	0.16	9,200	4.6
Hydrogen deflagration	1.1	Once in 18,000 years	1.0	0.11	6,400	3.2
Criticality	47,000	Once in 77,000 years	0.029	0.0012	18	0.009

a. Source: TtNUS (1999b).

b. MEI = Maximally exposed individual.

Table D-4. FB-Line radiological accidents and impacts.

Accident	Maximum curies released	Accident frequency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^a (rem)	Offsite population (person-rem)	Latent cancer fatalities
Design basis earthquake, 0.2g intensity ^b	0.31	Once in 5,000 years	0.34	0.042	150	0.077
Propagated fire ^c	2.2	Once in 59,000 years	0.18	0.14	1,100	0.53

a. MEI = Maximally exposed individual.

b. Source: WSRC (1993).

c. Source: WSRC (1995).

Cask Decontamination Waste Released to Environment (TS-3)

Casks would be washed at receipt and before shipping. The wash liquid probably would be collected in a sump or storage tank and released to the environment if sample results showed contamination levels to be within acceptable limits. Excessively radioactive or hazardous material could be pumped inadvertently to an outfall rather than to the liquid radioactive waste

system or hazardous waste storage if there was an error in processing samples or reading laboratory test results. This scenario assumes that happens and a small amount becomes airborne. The total curies released to air would be 2.0×10^{-7} and would result in negligible consequences. This event is postulated to occur once in 100 years of operation. The calculated consequences for this scenario are listed in Table D-7.

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Table D-5. Receiving Basin for Offsite Fuel radiological accidents and impacts.^a

Accident	Maximum curies released	Accident frequency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
Fuel rupture	160,000	Once in 1.4 years	0.0018	1.9×10^{-4}	10	0.005
Resin explosion	1.0	Once in 400 years	0.0012	1.3×10^{-4}	7.8	0.0039
Uncontrolled chemical reaction	1,600,000	Once in 450 years	0.018	0.0019	100	0.05
Resin fire	11	Once in 1,200 years	1.3×10^{-4}	1.4×10^{-5}	0.83	4.2×10^{-4}
Process-induced criticality	4,800	Once in 1,500 years	0.16	0.016	970	0.49
NPH ^d (high winds)						
Fuel breach	1,600,000	Once in 2,600 years	0.13	0.0024	130	0.063
Criticality	48,000	Once in 26,000 years	13	0.22	12,000	6.2
NPH (earthquake)						
Waste tank breach	0.69	Once in 280 years	0.0065	1.1×10^{-4}	6.3	3.2×10^{-3}
Fuel breach	1,600,000	Once in 36,000,000 years	0.13	0.0024	130	0.063
Criticality	48,000	Once in 38,000,000 years	13	0.22	12,000	6.2

TC

- a. Source: TtNUS (1999a).
b. MEI = Maximally exposed individual.
c. Data not available.
d. NPH = Natural Phenomenon Hazard.

Table D-6. Reactor Disassembly Basins radiological accidents and impacts.^a

Accident	Maximum curies released	Accident fre- quency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
Basin overfill	(c)	Once in 100 years	0	4.6×10^{-4}	(c)	
Mishandling fuel assembly	(c)	Once in 100 years	25	0	0	0
Criticality	4,800	Once in 300 years	0.16	0.016	660	0.3
Basin water draindown	(c)	Once in 500 years	0.055	0.016	(c)	

TC

- a. Source: WSRC (1997b), TtNUS (1999a).
b. MEI = Maximally exposed individual.
c. Accidents expected to result in low consequences and risks to the onsite worker population and the offsite population. Quantitative estimates of consequences and risks are not available.

TC

Table D-7. Transfer Facility radiological accidents and impacts.^a

Accident	Maximum curies released	Accident frequency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
Railroad car contamination, TS-1	1.0×10^{-7}	Once in 10 years	1.7×10^{-7}	1.8×10^{-8}	7.4×10^{-4}	3.7×10^{-7}
Sump Release, TS-3	2.1×10^{-7}	Once in 100 years	3.5×10^{-7}	3.6×10^{-8}	1.5×10^{-3}	7.4×10^{-7}

a. Source: TtNUS (1999a).
b. MEI = Maximally exposed individual.

Cask Criticality From Internal Disruption - Single Shipping Cask (TS-4)

Shipping casks containing spent fuel would be moved between facilities by rail or truck and loaded or unloaded from transports using overhead or mobile cranes. The casks contain internal structures that maintain fuel separation or provide neutron absorption. Casks certified in accordance with regulations of the U.S. Department of Transportation are designed to withstand drops from a specified height. However, if the cask internal structures were not assembled properly, disruption and redistribution of the fuel could occur. In addition, the fuel itself could be damaged or lose its integrity and redistribute itself in a critical configuration. This could produce a criticality at the time of the disruption or later if the cask were filled with water for purging. If this event occurred outdoors, the release would not be filtered. This event is not credible.

Criticality From Fuel Dropped On To Floor Or In To Dry Storage Rack (TS-5)

A criticality accident could result if spent nuclear fuel were dropped in a pile on the floor or dropped into the cask-unloading dry storage rack. The fuel drop could result from operator error or equipment failure in the handling machine or spent fuel structure. Double contingency protection would require the dropping of at least two spent fuel loads (assemblies, canisters, bundles, etc.) before a criticality occurred. In addition, the first drop would have to be unrecovered when the second drop occurred. Pro-

cedures would require the recovery of the first fuel dropped before material movement could resume. The effects of a criticality event would be mitigated by shielding and the physical distance of the operators in remote handling operations. This event would occur inside the facility, so released radionuclides would be filtered. This event is not considered credible.

Criticality of Spent Fuel in Several Adjacent Shipping Casks (TS-7)

This event addresses a criticality accident among spent nuclear fuel brought together in multiple shipping or storage casks. Because the nuclear reaction would occur in all the fuel in the array, several casks could be involved. A criticality accident of this nature would produce a direct radiation hazard and would release radioactive contamination if one or more casks were breached. However, a criticality during cask dry storage is not a credible event.

D.3.4.2 Dry Storage Facility Accidents

Spent Nuclear Fuel Dry Storage Process-Related Criticality Accident (SLS-2)

This accident scenario involves criticality resulting from the improper loading of dry storage racks or the mechanical failure of racks. Mechanical failure or collapse could result from a crane impact or structural flaw in the racks. Improper loading would result in sufficient spent nuclear fuel assemblies placed near one another with insufficient neutron absorbers to prevent a criticality. A collapse of the racks would result in sufficient spent fuel assemblies piled near one

TC | another in the debris with insufficient neutron absorbers to prevent a criticality. This event is postulated to occur once in 330 years of operation. The calculated consequences for this scenario are listed in Table D-8.

Natural Phenomenon Hazard-Induced Spent Nuclear Fuel Dry Storage Criticality Accident (SLS-3)

TC | This accident scenario involves a natural phenomenon hazard-induced criticality resulting from an earthquake-induced mechanical failure of racks (e.g., collapse or crane impact and collapse) or a subsequent fission product release resulting from a fuel breach. This event is predicated on the assumption that the facility could withstand an earthquake intact and operational. However, the scenario assumes that the structure that contains the material at risk fails, resulting in the event. This event is postulated to occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-8.

Spent Nuclear Fuel Dry Storage Fission Product Release (SLS-1)

TC | This accident scenario involves the release of fission products from a fuel breach and a simultaneous loss of confinement due to an earthquake. The fuel breach would result from an earthquake-induced mechanical failure or collapse of the storage racks or an earthquake-induced crane impact. This event is postulated to occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-8.

D.3.5 ELECTROMETALLURGICAL TREATMENT ACCIDENT SEQUENCES

Melter Eruption (GFP-1 and MM-1)

The electrometallurgical technology would have two separate melters. The melter eruption postulated event could result from impurities in the glass melt (GFP-1) or the metal melt (MM-1).

Impurities could range from water that could cause a steam eruption to chemical contaminants that could react at elevated temperatures and produce a highly exothermic reaction (eruption or deflagration). The scenario assumes that the resulting sudden pressure increase would eject the fissile-material-bearing melt liquid into the processing structure. It also assumes that the energy release would not damage the processing structure and its associated filtered exhaust ventilation system. The melter ventilation systems would remove or dilute explosive mixtures that could build up in the gas space above the molten material. Operating procedures and verifications would prevent the addition of impure or incorrect materials to the melt. Therefore, this event is postulated to occur once in 20 years of operation.

If the eruption was large, operating personnel would hear and see it. A small eruption might be detected only by airborne radiation monitors because the remotely-operated melters would be in a heavily shielded area. The effects of the eruption would be mitigated by the melter design, which would include venting methods to respond to an over-pressure event. The melter building structure and the ventilation system would confine particulate radioactive material released in the eruption. The calculated consequences for this scenario are listed in Table D-9 for the molten glass release and Table D-10 for the molten metal spill. Noble gases and tritium released on the event would not be filtered.

Earthquake-Induced Fission Product Release and Confinement Failure (GFP-4 and MM-5)

The fission-product-release scenario involves damage to the melter structure and its associated systems that would release fission products. This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the event. This event is postulated to occur once in 2,000 years of operation. The calculated

Accident Analysis

Table D-8. Dry Storage radiological accidents and impacts.^a

Accident	Maximum curies released	Accident frequency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
Criticality in storage, SLS-2	4,800	Once in 330 years	0.16	0.016	660	0.3
Earthquake-induced criticality, SLS-3	48,000	Once in 2,000 years	13	0.22	12,000	6.2
Fuel breach during earthquake, SLS-1	1,100,000	Once in 2,000 years	0.014	0.0015	54.1	0.027

a. Source: TtNUS (1999a).

b. MEI = Maximally expose individual.

Table D-9. Electrometallurgical Treatment radiological accidents and impacts (glass melter only).^a

Accident	Maximum curies released	Accident frequency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
Melter eruption, GFP-1	1,200	Once in 20 years	1.6×10^{-5}	1.1×10^{-6}	0.04	2.0×10^{-5}
Earthquake-induced fission product release and confinement failure, GFP-4	2,300	Once in 2,000 years	3.2×10^{-5}	2.2×10^{-6}	0.08	4.0×10^{-5}
Melter eruption with loss of venti- lation, GFP-1a	1,200	Once in 2,000 years	0.002	2.3×10^{-4}	9.5	0.0047
Earthquake spill with loss of ven- tilation, GFP-4a	2,300	Once in 200,000 years	0.038	6.2×10^{-4}	26	0.013

a. Source: TtNUS (1999a).

b. MEI = Maximally exposed individual.

Table D-10. Melt and Dilute Treatment radiological accidents and impacts (these accidents also apply to the metal melter for Electrometallurgical Treatment).^a

Accident	Maximum curies released	Accident frequency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
Melter eruption, MM-1	0.09	Once in 20 years	7.1×10^{-6}	7.4×10^{-7}	0.03	1.5×10^{-5}
Criticality due to multiple batching 5×10^{17} fissions, MM-4	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake-induced fission product release and confinement failure, MM-5	2,300	Once in 2,000 years	6.8×10^{-5}	5.9×10^{-6}	0.23	1.2×10^{-4}
Melter eruption with loss of venti- lation, MM-1a	9,200	Once in 2,000 years	0.71	0.074	3,000	1.5
Process criticality with loss of ven- tilation, MM-4a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake-induced spill with loss of ventilation, MM-5a	21,000	Once in 200,000 years	30	0.50	21,000	10

a. Source: TtNUS (1999a), TtNUS (2000).

b. MEI = Maximally exposed individual.

consequences for this scenario are listed in Table D-9 for the glass melt and Table D-10 for the metal melt.

Earthquake-Induced Fission Product Release and Confinement Failure (GFP-4 and MM-5)

The fission-product-release scenario involves damage to the melter structure and its associated systems that would release fission products. This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the event. This event is postulated to occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-9 for the glass melt and Table D-10 for the metal melt.

Criticality Accident (MM-4)

Melter design volume limits would prevent a criticality accident. However, to preserve flexibility of operation, there would have to be provision for some excess volume, so a criticality accident could result from charging multiple batches of fissile material to the metal melter. The batching operation would be a procedure-guided operator action. "Double Contingency" would require a second operator to verify the processing steps independently. Therefore, this event is postulated to occur once in 330 years of operation. In the event of a criticality, the process building structure and filtered exhaust system would remain intact and would confine fission products and shield against direct radiation exposure. The calculated consequences for this scenario are listed in Table D-10.

Melter Eruption with Coincident Ventilation Failure (GFP-1a and MM-1a)

This scenario has the same initiating event as the glass melt eruption (GFP-1) or the metal melt eruption (MM-1) but with a coincident failure of the HEPA filtration system. As this event requires both a melter eruption and a ventilation failure, the postulated frequency for this event is once in 2,000 years. The calculated

consequences for this scenario are listed in Table D-9 for the molten glass release and Table D-10 for the molten metal spill.

Earthquake-Induced Fission Product Release and Confinement Failure with Coincident Ventilation Failure (GFP-4a and MM-5a)

This scenario has the same initiating event as the glass melt spill (GFP-4) or the metal melt spill (MM-5) but with a coincident failure of the HEPA filtration system. As this event requires both a seismic event and a ventilation failure, the postulated frequency for this event is once in 200,000 years. The calculated consequences for this scenario are listed in Table D-9 for the molten glass release and Table D-10 for the molten metal spill.

Criticality Accident with Coincident Ventilation Failure (MM-4a)

This scenario has the same initiating event as the criticality accident (MM-4), but with a coincident failure of the HEPA filtration system. As this event requires both a criticality and a ventilation failure, the postulated frequency for this event is once in 33,000 years. The calculated consequences for this scenario are listed in Table D-10.

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.6 MELT AND DILUTE TREATMENT FACILITY ACCIDENT SEQUENCES

The accidents for the Melt and Dilute Treatment Facility would be the same for either a new facility or in a renovated reactor building.

Melter Eruption (MM-1)

This event would be identical to the metal melt eruption described as MM-1 in D.3.5. The cal-

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culated consequences are presented in Table D-10.

Criticality Accident (MM-4)

The criticality event would be identical to that described for the metal melter as MM-4 in D.3.5. The calculated consequences are presented in Table D-10.

Earthquake-Induced Fission Product Release and Confinement Failure (MM-5)

The fission product release and confinement failure would be identical to that described as MM-5 in D.3.5. The calculated consequences are presented in Table D-10.

Melter Eruption with Ventilation Failure (MM-1a)

This event would be identical to the metal melt eruption with ventilation failure described in D.3.5. The calculated consequences are presented in Table D-10.

Earthquake-Induced Fission Product Release and Confinement Failure with Coincident Ventilation Failure (MM-5a)

This event would be identical to the Earthquake-induced spill with ventilation failure described in D.3.5. The calculated consequences are presented in Table D-10.

Criticality Accident with Coincident Ventilation Failure (MM-4a)

This event would be identical to the Double Batching Criticality with ventilation failure described in D.3.5. The calculated consequences are presented in Table D-10.

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.7 MECHANICAL DILUTION TREATMENT

D.3.7.1 Press and Dilute Treatment Accident Sequences

Fission Product Release (SDP-2)

This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the event. This event is postulated to occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-11.

Spent Nuclear Fuel-Depleted Uranium Press Process Criticality Accident (SDP-3)

This process-related criticality would result from multiple batches of spent fuel plates introduced into the press or an inadvertent substitution of spent fuel plates for depleted uranium plates. In either instance sufficient spent fuel in the configuration would result in a criticality. This event is postulated to occur once in 330 years of operation. The calculated consequences for this scenario are listed in Table D-11.

Earthquake-Induced Fire/Pyrophoric Reaction (SDP-4)

An earthquake-induced fire or pyrophoric reaction would result from friction due to mechanical shredding, electrical or mechanically induced fires on uranium metal fuel, or a fire started by a hydraulic fluid leak that resulted in a subsequent pyrophoric reaction. This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the event. This event is postulated to occur once in 20,000 years of operation. The calculated consequences for this scenario are listed in Table D-11.

Table D-11. Press and Dilute radiological accidents and impacts.^a

Accident	Maximum curies released	Accident frequency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
SNF-DU press process criticality, SDP-3	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake induce fission product release, SDP-2	230	Once in 2,000 years	3.2×10^{-6}	2.2×10^{-7}	8.0×10^{-3}	4.0×10^{-6}
Earthquake-induced fire/pyrophoric reaction, SDP-4	2,300	Once in 20,000 years	3.6×10^{-5}	2.6×10^{-6}	0.095	4.8×10^{-5}
Process criticality with loss of ventilation, SDP-3a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake-induced fission product release with loss of ventilation, SDP-2a	240	Once in 200,000 years	0.010	1.6×10^{-4}	6.6	0.0033

a. Source: TtNUS (1999a).

b. MEI = Maximally exposed individual.

Fission Product Release with Coincident Ventilation Failure (SDP-2a)

This scenario has the same initiating event as the Fission Product Release Accident (SDP-2), but with a coincident failure of the HEPA filtration system. As this event requires both an earthquake and a ventilation failure, the postulated frequency for this event is once in 200,000 years. The calculated consequences for this scenario are listed in Table D-11.

Spent Nuclear Fuel-Depleted Uranium Press Process Criticality Accident (SDP-3a)

This scenario has the same initiating event as the criticality accident (SDP-3), but with a coincident failure of the HEPA filtration system. As this event requires both a criticality and a ventilation failure, the postulated frequency for this event is once in 33,000 years. The calculated consequences for this scenario are listed in Table D-11.

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.7.2 Chop and Dilute Treatment Accident Sequences

Fission Product Release (SS-2)

This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the event. This event is postulated to occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-12.

Spent Nuclear Fuel Shredding Process Criticality Accident (SS-3)

This process-related criticality would result from feeding multiple batches of spent fuel to the fuel shredder. This would result in sufficient spent fuel in a configuration that could result in a criticality. This event is postulated to occur once in 330 years of operation. The calculated consequences for this scenario are listed in Table D-12.

Earthquake-Induced Fire/Pyrophoric Reaction (SS-4)

An earthquake-induced fire or pyrophoric reaction could result from friction due to mechanical

Table D-12. Chop and Dilute radiological accidents and impacts.^a

Accident	Maximum curies released	Accident frequency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
Process criticality, SS-3	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake-induced fission product release, SS-2	0.07	Once in 2,000 years	1.2×10^{-7}	1.2×10^{-8}	4.9×10^{-4}	2.5×10^{-7}
Earthquake-induced fire, SS-4	2.3	Once in 2,000 years	3.6×10^{-6}	3.8×10^{-7}	0.015	7.7×10^{-6}
Process criticality with loss of ventilation, SS-3a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake release with loss of ventilation, SS-2a	66	Once in 200,000 years	0.012	0.0012	49	0.024
Earthquake-induced fire with loss of ventilation, SS-4a	2,100	Once in 200,000 years	3	0.050	2,100	1.0

a. Source: TtNUS (1999a).

b. MEI = Maximally exposed individual.

shredding, electrical or mechanically induced fires on uranium metal fuel, or a fire started by a hydraulic fluid leak that resulted in a subsequent pyrophoric reaction. This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the event. This event is postulated to occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-12.

Fission Product Release with Coincident Ventilation Failure (SS-2a)

This scenario has the same initiating event as the Fission Product Release Accident (SS-2), but with a coincident failure of the HEPA filtration system. As this event requires both an earthquake and a ventilation failure, the postulated frequency for this event is once in 200,000 years. The calculated consequences for this scenario are listed in Table D-12.

Spent Nuclear Fuel Shredding Process Criticality Accident (SS-3a)

This scenario has the same initiating event as the criticality accident (SS-3), but with a coincident failure of the HEPA filtration system.

As this event requires both a criticality and a ventilation failure, the postulated frequency for this event is once in 33,000 years. The calculated consequences for this scenario are listed in Table D-12.

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.8 VITRIFICATION FACILITIES

D.3.8.1 Glass Material Oxidation and Dissolution System Accident Scenarios

Melter Eruption (GMF-1)

The postulated melter eruption could result from impurities in the metal melt. Impurities could range from water that could cause a steam eruption to chemical contaminants that could react at elevated temperatures and produce a highly exothermic reaction (eruption or deflagration). This scenario assumes that the resulting sudden pressure increase would eject the fissile-material-bearing melt liquid into the processing structure. It also assumes that the energy release

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would not damage the processing structure and its associated filtered exhaust ventilation system. The melter offgas and inerting systems would remove or dilute explosive mixtures that might build up in the gas space above the molten material. Operating procedures and verifications prevent the addition of impure or incorrect materials to the melt. Therefore, this event is postulated to occur once in 20 years of operation.

If a large eruption did occur, the appearance and sound would alert operating personnel. A small eruption might be detected only by airborne radiation monitors because the remotely operated melters would be in a heavily shielded area. The effects of the eruption would be mitigated by the design of the melter, which would include venting methods to respond to an overpressure event. The melter building structure and the ventilation system would confine particulate radioactive material released in the eruption. The calculated consequences for this scenario are listed in Table D-13.

Criticality Accident (GMF-4)

Melter design volume limits would prevent a criticality accident. However, to preserve flexibility of operation, there would have to be pro-

vision for some excess volume, a criticality accident could result from charging multiple batches of fissile material to the metal melter. The batching operation would be a procedure-guided operator action. "Double Contingency" would require a second operator to verify the processing steps independently. Therefore, this event is postulated to occur once in 33,000 years of operation. In the event of a criticality, the process building structure and filtered exhaust system would remain intact and would confine fission products and shield against direct radiation exposure. The calculated consequences for this scenario are listed in Table D-13.

Earthquake-Induced Fission Product Release and Confinement Failure (GMF-5)

The fission-product-release scenario involves damage to the melter structure and its associated systems that would release fission products. This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the release. This event is postulated to occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-13.

Table D-13. GMODS radiological accidents and impacts.^a

Accident	Maximum curies released	Accident frequency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
Melter eruption, GMF-1	1,200	Once in 20 years	1.6×10^{-5}	1.1×10^{-6}	0.04	2.0×10^{-5}
Criticality due to multiple batching 5×10^{17} fissions, GMF-4	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake-induced fission product release and confinement failure, GMF-5	2,300	Once in 2,000 years	3.3×10^{-5}	2.2×10^{-6}	0.08	4.0×10^{-5}
Melter eruption with loss of venti- lation, GMF-1a	1,200	Once in 2,000 years	0.0024	2.6×10^{-4}	10	0.0052
Process criticality with loss of ven- tilation, GMF-4a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake-induced release with loss of ventilation, GMF-5a	2,300	Once in 200,000 years	0.041	6.8×10^{-4}	28	0.014

a. Source: TtNUS (1999a).

b. MEI = Maximally exposed individual.

TC

Melter Eruption with Coincident Ventilation Failure (GMF-1a)

This scenario has the same initiating event as the melter eruption (GMF-1), but with a coincident failure of the HEPA filtration system. As this event requires both a melter eruption and a ventilation failure, the postulated frequency for this event is once in 2,000 years. The calculated consequences for this scenario are listed in Table D-13.

Earthquake-Induced Fission Product Release and Confinement Failure with Coincident Ventilation Failure (GMF-4a)

This scenario has the same initiating event as the earthquake-induced melt spill (GMF-4), but with a coincident failure of the HEPA filtration system. As this event requires both a seismic event and a ventilation failure, the postulated frequency for this event is once in 200,000 years. The calculated consequences for this scenario are listed in Table D-13.

Criticality Accident with Coincident Ventilation Failure (GMF-4a)

This scenario has the same initiating event as the criticality accident (GMF-4), but with a coincident failure of the HEPA filtration system. As this event requires both a criticality and a ventilation failure, the postulated frequency for this event is once in 33,000 years. The calculated consequences for this scenario are listed in Table D-13.

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.8.2 Plasma Arc Accident Scenarios

Melter Eruption (GMF-1), Criticality Accident (GMF-4), and Earthquake-Induced Fission Product Release and Confinement Failure (GMF-5) and Corresponding Events with Coincident Loss of Ventilation (GMF-1a, 4a, and 5a)

These events are identical in description to GMF-1, GMF-1a, GMF-4, GMF-4a, GMF-5, and GMF-5a as described in D.3.8.1 for the Glass Material Oxidation and Dissolution System accident scenarios. The calculated consequences for these scenarios are listed in Table D-14.

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.8.3 F Canyon Dissolve and Vitrify Treatment Accident Sequences

Melter Eruption (GMF-1), Criticality Accident (GMF-4), and Earthquake-Induced Fission Product Release and Confinement Failure (GMF-5) and Corresponding Events with Coincident Loss of Ventilation (GMF-1a, 4a, and 5a)

These events are identical in description to GMF-1, GMF-1a, GMF-4, GMF-4a, GMF-5, and GMF-5a as described in D.3.8.1 for the Glass Material Oxidation and Dissolution System accident scenarios. The calculated consequences for these scenarios are listed in Table D-15.

TC

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Table D-14. Plasma Arc radiological accidents and impacts.^a

Accident	Maximum curies released	Accident fre- quency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
Melter eruption, GMF-1	1,200	Once in 20 years	1.6×10^{-5}	1.1×10^{-6}	0.040	2.0×10^{-5}
Criticality due to multiple batching 5×10^{17} fissions, GMF-4	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake-induced fission product release and confinement failure, GMF-5	2,300	Once in 2,000 years	3.3×10^{-5}	2.2×10^{-6}	0.080	4.0×10^{-5}
Melter eruption with loss of venti- lation, GMF-1a	1,200	Once in 2,000 years	0.0062	6.4×10^{-4}	26	0.013
Process criticality with loss of ven- tilation, GMF-4a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake-induced release with loss of ventilation, GMF-5a	2,400	Once in 200,000 years	0.10	0.0017	71	0.035

a. Source: TtNUS (1999a).
b. MEI = Maximally exposed individual.

Table D-15. F Canyon Dissolve and Vitrify radiological accidents and impacts.^a

Accident	Maximum curies released	Accident frequency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
Glass melt eruption, GMF-1	1,200	Once in 20 years	1.3×10^{-5}	1.2×10^{-6}	0.044	2.2×10^{-5}
Criticality due to multiple batching, GMF-4	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake-induced fission product release and confinement failure, GMF-5	2,300	Once in 2,000 years	2.5×10^{-5}	2.4×10^{-6}	0.088	4.4×10^{-5}
Glass melt eruption with loss of ventilation, GMF-1a	1,200	Once in 2,000 years	0.0019	2.8×10^{-4}	11	0.0056
Process criticality with loss of ven- tilation, GMF-4a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake-induced release with loss of ventilation, GMF-5a	2,300	Once in 200,000 years	0.051	8.1×10^{-4}	32	0.016

a. Source: TtNUS (1999a).
b. MEI = Maximally exposed individual.

Accident Analysis**Other Accident Scenarios**

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.8.4 New Dissolve and Vitrify Facility

Melter Eruption (GMF-1), Criticality Accident (GMF-4) and Earthquake-Induced Fission Product Release and Confinement Failure (GMF-5) and Corresponding Events with Coincident Loss of Ventilation (GMF-1a, 4a, and 5a)

These events are identical in description to GMF-1, GMF-1a, GMF-4, GMF-4a, GMF-5, and GMF-5a as described in D.3.8.1 for the Glass Material Oxidation and Dissolution System accident scenarios. The calculated consequences for these scenarios are listed in Table D-16.

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not

listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.4 Comparison of Accident Impacts for Alternative Facility Locations

An alternative location for new facilities would be C-Area. If DOE located the facilities in C-Area, doses to the MEI and the population would be expected to be approximately 4.0 percent and 12 percent higher, respectively, due to the shorter distance to the Site boundary.

D.5 Impacts of Postulated Accidents Involving Nonradioactive Hazardous Materials

This section summarizes the impacts of potential accidents involving hazardous chemicals at the Receiving Basin for Offsite Fuel as documented in the safety analysis report for the facility (WSRC 1995). These accidents would not involve radioactive materials.

Table D-16. New Dissolve and Vitrify radiological accidents and impacts.^a

Accident	Maximum curies released	Accident frequency	Accident Consequences			
			Noninvolved worker (rem)	MEI ^b (rem)	Offsite population (person-rem)	Latent cancer fatalities
Glass melt eruption, GMF-1	1,200	Once in 20 years	1.6×10^{-5}	1.1×10^{-6}	0.04	2.0×10^{-5}
Criticality due to multiple batching GMF-4	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake-induced fission product release and confinement failure, GMF-5	2,300	Once in 2,000 years	3.3×10^{-5}	2.2×10^{-6}	0.08	4.0×10^{-5}
Melter eruption with loss of ventilation, GMF-1a	1,200	Once in 2,000 years	0.0024	2.6×10^{-4}	10	0.0052
Process criticality with loss of ventilation, GMF-4a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake-induced release with loss of ventilation GMF-5a	2,300	Once in 200,000 years	0.041	6.8×10^{-4}	28	0.014

a. Source: TtNUS (1999a).

b. MEI = Maximally exposed individual.

The hazard analysis documented in the safety analysis report identified three chemical spill events that required unique accident analyses. This section describes the analysis of these events, which include chemical spills of sodium hydroxide, nitric acid, and sodium nitrite from storage dumpsters outside the facility.

D.5.1 LOSS OF 50-PERCENT SODIUM HYDROXIDE CONTAINMENT

Sodium hydroxide (NaOH), used for anion resin regeneration, is stored in a skid-mounted 1,000-gallon dumpster on a chemical pad west of Building 245-H. This dumpster is typically filled to 900 gallons and is heated during the winter to approximately 10 to 12°F above the crystallization point using 25-psi steam routed through piping inside the dumpster.

If an initiating event occurred that ruptured the tank, the chemical would accumulate in the bermed area of the pad. The rate of leakage from the dumpster would depend on the point of the breach and the severity of the opening. A worst-case breach would drain the contents of the dumpster within minutes. This scenario takes no credit for the berm containing the chemical spill. Therefore, the sodium hydroxide would spread over a large area, which would result in a larger airborne release rate than would a bermed release.

The sodium hydroxide plume would be transported by the wind as tiny particles. Therefore, a Gaussian plume model is appropriate. This event is postulated to occur once in 190 years of operation.

The calculated concentration would be lower than the lowest concentration guideline (PEL-TWA) for either on- or offsite. Therefore, the consequences of the release would be insignificant and there is no need for further analysis at greater distances.

D.5.2 LOSS OF 50-PERCENT NITRIC ACID CONTAINMENT

DOE uses nitric acid (HNO₃) in the regeneration of cation resin, and stores it in a skid-mounted 1,000-gallon dumpster west of Building 245-H. The dumpster is typically filled to 900 gallons. Nitric acid is supplied to the Resin Regeneration Facility through underground piping. The chemical pad is approximately at ground level outside the Receiving Basin for Offsite Fuel and the Resin Regeneration Facility. It is surrounded by a dike to contain spills.

If an initiating event occurred that ruptured the tank, the chemical would accumulate in the bermed area of the pad. The rate of leakage from the dumpster would depend on the point of the breach and the severity of the opening. A worst-case breach would drain the contents of the dumpster within minutes. This scenario takes no credit for the berm containing the chemical spill. Therefore, the nitric acid would spread over a large area, which would result in a larger airborne release rate than would a bermed release. This event is postulated to occur once in 190 years of operation.

The release would result in a concentration of 3.1×10^{-3} milligrams per cubic meter at 640 meters (2,100 feet) and 4.0×10^{-4} milligrams per cubic meter at the nearest Site boundary. These values are both lower than the Emergency Response Planning Guideline-2 values.

D.5.3 LOSS OF 30-PERCENT SODIUM NITRITE CONTAINMENT

DOE stores sodium nitrite (NaNO₂), a waste tank corrosion inhibitor, in a skid-mounted 1,000-gallon dumpster on a chemical pad west of Building 245-H. The contents of the dumpster are pumped to an adjacent Holdup Tank with a maximum capacity of 1,600 gallons. This analysis assumes that the contents of both

tanks are filled to their total combined volume of 2,600 gallons. The chemical pad is approximately at ground level outside the Receiving Basin for Offsite Fuel and the Resin Regeneration Facility, and is surrounded by a dike.

If an initiating event occurred that ruptured the tank, the chemical would accumulate in the bermed area of the pad. The rate of leakage from the dumpster or holdup tank would depend on the point of the breach and the severity of the opening. A worst-case breach would drain the contents of the dumpster or holdup tank within minutes. This scenario takes no credit for the berm containing the chemical spill. Therefore, the sodium nitrite would spread over a large area, which would result in a larger airborne release rate than would a bermed release. This event is postulated to occur once in 180 years of operation.

The calculated airborne concentration at a downwind distance of 100 meters (328 feet) would be 0.006 milligrams per cubic meter, which is less than the lowest concentration guideline (PEL-TWA). Therefore, the consequences of the release would be insignificant, and there is no need for analysis at greater distances.

D.5.4 SURFACE VEHICLE IMPACT

The impact of a surface vehicle with a chemical dumpster has been identified as a potential initiating event for chemical leakage. The consequences of the events would be the same as the consequences for the events analyzed in Sections D.5.1 through D.5.3. The postulated frequency for each of these chemical releases from surface vehicle impact would be once in 3,400 years.

D.6 Environmental Justice

In the event of an accidental release of radioactive or hazardous chemical substances, the dispersion of such substances would depend on meteorology conditions, such as wind direction, at the time. Given the variability of meteorology conditions and the low probability and risk of accidents, an accident would be unlikely to occur that would result in disproportionately high or adverse human health and environmental impacts to minorities or low-income populations.

References

- AIHA (American Industrial Hygiene Association), 1991, *Emergency Response Planning Guideline*, Emergency Response Planning Guidelines Committee, Akron, Ohio.
- DOE (U.S. Department of Energy), 1994, *Preparation Guide for U.S. DOE Nonreactor Nuclear Facility Safety Analysis Reports*, Standard DOE-STD-3009-94, Washington, D.C., February. | EC
- NCRP (National Council on Radiation Protection and Measurements), 1993, *Limits of Exposure to Ionizing Radiation*, Report No. 116, Bethesda, Maryland. | EC
- NRC (U.S. Nuclear Regulatory Commission), 1982, *Regulatory Guide 1.145: Atmospheric Dispersion Models for Potential Accidental Consequence Assessments at Nuclear Power Plants*, Revision 1, Washington D.C.
- Simpkins, A. A., 1995a, *Verification of AXAIRQ*, WSRC-RP-95-708, Westinghouse Savannah River Company, Savannah River Technology Center, Aiken, South Carolina.
- Simpkins, A. A., 1995b, *Verification of AXAIRQ*, WSRC-RP-95-709, Westinghouse Savannah River Company, Savannah River Technology Center, Aiken, South Carolina.
- Simpkins, A. A., 1997, memorandum to C. B. Shedrow, "Spent Nuclear Fuel EIS Accidental Release Environmental Dosimetry Calculations," SRT-EST-97-196, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina, March 13.
- Tetra Tech NUS, Inc. (TtNUS), 1999a, *Data Package Calculation of Accident Consequences for the Spent Nuclear Fuel EIS TTNUS-99-009*, Revision 0, TtNUS, Aiken, South Carolina, September 24. | TC
- Tetra Tech NUS, Inc. (TtNUS), 1999b, *Data Package Calculation of Accident Consequences for the Spent Nuclear Fuel EIS in H-Canyon TTNUS-99-010*, Revision 0, TtNUS, Aiken, South Carolina, October 14. | TC
- Tetra Tech NUS, Inc. (TtNUS), 2000, *Addendum to Data Package Calculation of Accident Consequences for the Spent Nuclear Fuel EIS*, TTNUS-99-009, Addendum, TtNUS, Aiken, South Carolina, January 28. | TC
- WSRC (Westinghouse Savannah River Company), 1993, *Calculation Note SRL-SEP-93-9010, FB-Line Basis for Interim Operations*, WSRC-RP-93-1102, Revision 0, Savannah River Site, Aiken, South Carolina.
- WSRC (Westinghouse Savannah River Company), 1995, *Revised Consequences of Propagated Fire in FB-Line*, S-CLC-F-00173, Revision 0, Savannah River Site, Aiken, South Carolina.
- WSRC (Westinghouse Savannah River Company), 1997a, *Receiving Basin for Off-Site Fuel and Resin Regeneration Facility Safety Analysis Report*, WSRC-SA-11, Savannah River Site, Aiken, South Carolina, August.
- WSRC (Westinghouse Savannah River Company), 1997b, *Basis for Interim Operation (BIO) for the L-Reactor Facility*, WSRC-TR-95-0054, Revision 0, Savannah River Site, Aiken, South Carolina, December.

WSRC (Westinghouse Savannah River Company), 1997c, *Determination of Accident Sequence Selection and Source Term Selections for the Spent Nuclear Fuel Environmental Impact Statement*, S-CLC-G-00145, Revision 0, Savannah River Site, Aiken, South Carolina.

WSRC (Westinghouse Savannah River Company), 1998, *Preliminary Hazards Analysis for the Spent Nuclear Fuel Treatment and Storage Facility at Conceptual Design Phase (U)*, WSRC-TR-98-00450, Revision 0, Savannah River Site, Aiken, South Carolina.

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APPENDIX E

ASSUMED SPENT NUCLEAR FUEL MANAGEMENT ACTIVITY DURATIONS FOR ENVIRONMENTAL IMPACT ANALYSIS

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APPENDIX E. ASSUMED SPENT NUCLEAR FUEL MANAGEMENT ACTIVITY DURATIONS FOR ENVIRONMENTAL IMPACT ANALYSIS

This appendix presents the assumed durations for each spent nuclear fuel management activity necessary to implement the alternatives described in Chapter 2. DOE used these assumed durations to analyze the environmental impacts of these activities, as described in Chapter 4. These durations are used in calculating the total impacts for the following technical disciplines: worker and public health, waste generation, and utilities and energy consumption. To estimate total impacts, DOE identified the activities (phases) necessary to implement each technology, the amount of time required for each phase of the technology, and the annual impacts estimated to occur from each phase. DOE summed the annual impacts over the entire duration of each phase of a particular technology to determine the impacts of each phase, then summed the impacts of all the phases needed to implement that technology.

In estimating these durations, DOE assumed that implementation of the spent nuclear fuel management activity began in 1998 and that the final phase would end in 2035 (for a 38-year period of analysis). The years in which each technology is likely to be available are listed in Chapter 2. DOE assumed that wet storage would continue through the date that the technology was available. The projected environmental impacts of the treatment options would (on an annual basis) be greater than continued storage; assuming that wet storage would end when treatment became available is conservative. For Conventional Processing, DOE used historic data for F- and H-Canyon operations to estimate the time needed to process the entire inventory of each type of fuel. These durations (McWhorter 1997) are expressed in terms of "dissolver-years" (i.e., the time it would take to

process a given fuel group using only a single canyon dissolver). However, DOE might choose to process a given fuel group using two dissolvers concurrently. In that case, the actual duration would be half that listed in McWhorter (1997), but the annual environmental impact was estimated to be twice that of a single dissolver.

For every other technology (other than Continued Wet Storage), DOE used engineering judgment to estimate the duration of the treatment phase needed to treat the entire inventory for which the technology is applicable. DOE assumed that each new treatment technology would be capable of treating the entire spent nuclear fuel inventory in 7 years based on best engineering judgments of treatment rates. Similarly, DOE assumed that the activities necessary under the Direct Disposal/Direct Co-Disposal and Repackage and Prepare to Ship technologies (characterization, fuel conditioning, cropping, etc.) also would take 7 years for the entire inventory. DOE then assumed that the fraction of the total duration attributable to each fuel type would be equal to the fraction of that fuel type's fissile mass to the total fissile mass of spent nuclear fuel in the scope of this EIS. Use of fissile mass to calculate relative treatment durations is appropriate because it approximates the total radioactivity for each fuel group. Table E-1 lists these fissile mass fractions. Tables E-2 through E-9 list the assumed durations for each phase of the eight technologies analyzed in this EIS.

After treatment, DOE assumed that the treated and packaged fuel would be put in dry storage for the duration of the 38-year period of analysis.

Table E-1. Percent of total fissile mass attributable to each fuel type^a.

Fuel group	Percent of total fissile mass (%)
A. Uranium and thorium metal fuels	1.5
B. Material test reactor-like fuels	70
C. HEU/LEU oxides and silicides requiring resizing or special packaging	19
D. Loose uranium oxide in cans	4
E. Higher actinide targets	0.6
F. Non-aluminum-clad fuels	5

a. Source: Bickford et al. (1997).

Table E-2. Durations for Prepare for Direct Disposal/Direct Co-Disposal technology.

Fuel group	Wet storage duration (years)	Treatment duration (years) ^a	Dry storage duration (years)
A. Uranium and thorium metal fuels	10	0.11	27.9
B. Material test reactor-like fuels	10	5.43	22.6
C. HEU/LEU oxides and silicides requiring re- sizing or special packaging	10	1.46	26.5
D. Loose uranium oxide in cans	NA	NA	NA
E. Higher actinide targets	NA	NA	NA
F. Non-aluminum-clad fuels	NA	NA	NA

NA = Technology is not applicable to this fuel type.

a. Activities performed to prepare the fuel for direct disposal/direct co-disposal.

Table E-3. Durations for Repackage and Prepare to Ship technology.

Fuel group	Wet storage duration (years)	Treatment duration (years) ^a	Dry storage duration (years)
A. Uranium and thorium metal fuels	NA	NA	NA
B. Material test reactor-like fuels	NA	NA	NA
C. HEU/LEU oxides and silicides requiring re- sizing or special packaging	NA	NA	NA
D. Loose uranium oxide in cans	NA	NA	NA
E. Higher actinide targets	10	0.04	28
F. Non-aluminum-clad fuels	10	0.35	27.65

NA = Technology is not applicable to this fuel type.

a. Activities performed to prepare the fuel for offsite shipment.

Table E-4. Durations for Melt and Dilute technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)	Dry storage duration (years)
A. Uranium and thorium metal fuels	10	0.11	27.9
B. Material test reactor-like fuels	10	5.2	22.8
C. HEU/LEU oxides and silicides requiring re- sizing or special packaging	10	1.39	26.6
D. Loose uranium oxide in cans	10	0.29	27.7
E. Higher actinide targets	NA	NA	NA
F. Non-aluminum-clad fuels	NA	NA	NA

NA = Technology is not applicable to this fuel type.

Table E-5. Durations for Mechanical Dilution technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)	Dry storage duration (years)
A. Uranium and thorium metal fuels	NA	NA	NA
B. Material test reactor-like fuels	10	5.52	22.5
C. HEU/LEU oxides and silicides requiring re- sizing or special packaging	10	1.48	26.5
D. Loose uranium oxide in cans	NA	NA	NA
E. Higher actinide targets	NA	NA	NA
F. Non-aluminum-clad fuels	NA	NA	NA

NA = Technology is not applicable to this fuel type.

Table E-6. Durations for Vitrification Technologies technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)	Dry storage duration (years)
A. Uranium and thorium metal fuels	10	0.11	27.9
B. Material test reactor-like fuels	10	5.2	22.8
C. HEU/LEU oxides and silicides requiring re- sizing or special packaging	10	1.39	26.6
D. Loose uranium oxide in cans	10	0.29	27.7
E. Higher actinide targets	NA	NA	NA
F. Non-aluminum-clad fuels	NA	NA	NA

NA = Technology is not applicable to this fuel type.

Table E-7. Durations for Electrometallurgical Treatment technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)	Dry storage duration (years)
A. Uranium and thorium metal fuels	10	0.11	27.9
B. Material test reactor-like fuels	10	5.2	22.8
C. HEU/LEU oxides and silicides requiring re-sizing or special packaging	10	1.39	26.6
D. Loose uranium oxide in cans	10	0.29	27.7
E. Higher actinide targets	NA	NA	NA
F. Non-aluminum-clad fuels	NA	NA	NA

NA = Technology is not applicable to this fuel type.

Table E-8. Durations for Conventional Processing technology.

Fuel group	Wet storage duration (years)	Treatment duration (years) ^{a,b}	Dry storage duration (years) ^c
A. Uranium and thorium metal fuels	9	0.2	1
B. Material test reactor-like fuels	9	14.9	1
C. HEU/LEU oxides and silicides requiring re-sizing or special packaging	9	7.5	1
D. Loose uranium oxide in cans	9	2.2	1
E. Higher actinide targets	NA	NA	NA
F. Non-aluminum-clad fuels	NA	NA	NA

NA = Technology is not applicable to this fuel type.

- a. Durations represent active processing time and do not include downtimes normally associated with processing activities.
- b. Duration assumes only a single dissolver is used. If two dissolvers were used, the duration would be decreased by one-half.
- c. Indicates storage of resulting low enriched uranium awaiting sale.

Table E-9. Durations for Continued Wet Storage technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)	Dry storage duration (years)
A. Uranium and thorium metal fuels	38	NA	NA
B. Material test reactor-like fuels	38	NA	NA
C. HEU/LEU oxides and silicides requiring re-sizing or special packaging	38	NA	NA
D. Loose uranium oxide in cans	38	NA	NA
E. Higher actinide targets	38	NA	NA
F. Non-aluminum-clad fuels	38	NA	NA

NA = Not applicable.

References

Bickford, W. E., J. F. Krupa, D. L. McWhorter, M. E. Dupont, 1997, *Savannah River Site Spent Nuclear Fuel Environmental Impact Statement Engineering Data Book for Routine Releases*, WSRC-TR-97-0044, Westinghouse Savannah River Company, Aiken, South Carolina, April 8.

McWhorter, D. L., 1997, "SRS SNF Management EIS, SNF Processing Durations (U)," interoffice memorandum to C. B. Shedrow, Westinghouse Savannah River Company, Aiken, South Carolina, May 28.

APPENDIX F

ESTIMATED INCREMENTAL NONRADIOLOGICAL AIR CONCENTRATIONS ATTRIBUTABLE TO SPENT NUCLEAR FUEL MANAGEMENT ACTIVITIES

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Table F-1. Estimated maximum incremental concentrations of nonradiological air pollutants for noninvolved worker - Uranium and Thorium Metal Fuels (Fuel Group A).^a

Pollutant	Averaging time	Regulatory standard ^b	Incremental concentration for technology option ^c							
			1	2	3	4	5	6	7	8
Toxic pollutants (mg/m³)										
Nitric acid	24-hour	5	<0.01	NA	<0.01	NA	2.61	<0.01	2.61	<0.01
1,1,1-trichloroethane	24-hour	1,900	–	NA	–	NA	0.02	–	0.02	–
Benzene	24-hour	3.19	–	NA	–	NA	0.02	–	0.02	–
Ethanolamine	24-hour	6	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	435	–	NA	–	NA	<0.01	–	<0.01	–
Ethylene glycol	24-hour	None	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	0.75	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	80	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	0.2	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	1,800	<0.01	NA	<0.01	NA	0.02	<0.01	0.02	<0.01
Manganese	24-hour	5	–	NA	–	NA	<0.01	–	<0.01	–
Mercury	24-hour	0.1	–	NA	–	NA	–	–	<0.01	–
Methyl alcohol	24-hour	260	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	590	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	410	–	NA	–	NA	<0.01	–	<0.01	–
Methylene chloride	24-hour	86.7	–	NA	–	NA	0.02	–	0.02	–
Napthalene	24-hour	50	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	19	–	NA	–	NA	<0.01	–	<0.01	–
Phosphorus	24-hour	0.1	–	NA	–	NA	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	2.0	–	NA	–	NA	<0.01	–	<0.01	–
Toluene	24-hour	754	<0.01	NA	<0.01	NA	0.02	<0.01	0.02	<0.01
Trichloroethene	24-hour	537	–	NA	–	NA	<0.01	–	<0.01	–
Vinyl acetate	24-hour	None	–	NA	–	NA	<0.01	–	<0.01	–
Xylene	24-hour	435	<0.01	NA	<0.01	NA	0.03	<0.01	0.03	<0.01
Criteria pollutants (µg/m³)										
Nitrogen oxides	Annual	NA	0.02	NA	0.02	NA	36.34	0.02	36.34	–
Total suspended particulates (total dust)	24-hour	15	<0.01	NA	<0.01	NA	0.33	<0.01	0.33	–
Particulate matter (respirable fraction)	Annual	5	0.03	NA	0.03	NA	0.02	0.03	0.02	–
	24-hour	NA	0.33	NA	0.33	NA	0.20	0.33	0.20	–
Carbon monoxide	8-hour	55	0.08	NA	0.12	NA	1.57	0.12	1.57	<0.01
	1-hour	NA	0.26	NA	0.39	NA	4.89	0.39	4.89	<0.01
Sulfur dioxide	Annual	NA	<0.01	NA	<0.01	NA	0.02	<0.01	0.02	–
	8-hour	13	<0.01	NA	<0.01	NA	0.28	<0.01	0.28	–
	3-hour	NA	<0.01	NA	<0.01	NA	0.68	<0.01	0.68	–
Gaseous fluorides	1-month	None	–	NA	–	NA	0.10	–	0.10	–
	1-week	NA	–	NA	–	NA	0.17	–	0.17	–
	24-hour	NA	–	NA	–	NA	0.52	–	0.52	–
	12-hour	NA	–	NA	–	NA	0.76	–	0.76	–
Ozone (as VOC)	1-hour	0.2	NC	NA	NC	NA	NC	NC	NC	–

NA = Technology is not applicable to this fuel type.

— = No air emission associated with this option.

NC = Not Calculated.

VOC = volatile organic compound.

- a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.
- b. NIOSH (1991) and OSHA TWAs.
- c. Technology options: 1 = Prepare for Direct Disposal/Direct Co-Disposal; 2 = Repackage and Prepare to Ship; 3 = Melt and Dilute; 4 = Mechanical Dilution; 5 = Vitrification Technologies; 6 = Electrometallurgical Treatment; 7 = Conventional Processing; and 8 = Continued Wet Storage.

Table F-2. Estimated maximum incremental concentrations of nonradiological air pollutants for noninvolved worker - Materials Test Reactor-Like Fuels (Fuel Group B).^a

Averaging Regulatory			Incremental concentration for technology option ^c							
Pollutant	time	standard ^b	1	2	3	4	5	6	7	8
Toxic pollutants (mg/m ³)										
Nitric acid	24-hour	5	<0.01	NA	<0.01	<0.01	3.91	<0.01	3.91	<0.01
1,1,1-trichloroethane	24-hour	1,900	–	NA	–	–	0.02	–	0.02	–
Benzene	24-hour	3.19	–	NA	–	–	0.03	–	0.03	–
Ethanolamine	24-hour	6	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	435	–	NA	–	–	<0.01	–	<0.01	–
Ethylene glycol	24-hour	None	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	0.75	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	80	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	0.2	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	1,800	<0.01	NA	<0.01	<0.01	0.03	<0.01	0.03	<0.01
Manganese	24-hour	5	–	NA	–	–	<0.01	–	<0.01	–
Mercury	24-hour	0.1	–	NA	–	–	–	–	<0.01	–
Methyl alcohol	24-hour	260	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	590	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	410	–	NA	–	–	<0.01	–	<0.01	–
Methylene chloride	24-hour	86.7	–	NA	–	–	0.03	–	0.03	–
Napthalene	24-hour	50	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	19	–	NA	–	–	<0.01	–	<0.01	–
Phosphorus	24-hour	0.1	–	NA	–	–	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	2.0	–	NA	–	–	<0.01	–	<0.01	–
Toluene	24-hour	754	<0.01	NA	<0.01	<0.01	0.03	<0.01	0.03	<0.01
Trichloroethene	24-hour	537	–	NA	–	–	<0.01	–	<0.01	–
Vinyl acetate	24-hour	None	–	NA	–	–	<0.01	–	<0.01	–
Xylene	24-hour	435	<0.01	NA	<0.01	<0.01	0.05	<0.01	0.05	<0.01
Criteria pollutants (µg/m ³)										
Nitrogen oxides	Annual	NA	0.02	NA	0.04	0.02	54.51	0.04	54.51	–
Total suspended particulates (total dust)	24-hour	15	<0.01	NA	<0.01	<0.01	0.49	<0.01	0.49	–
Particulate matter (respirable fraction)	Annual	5	0.04	NA	0.04	0.04	0.03	0.04	0.03	–
	24-hour	NA	0.49	NA	0.49	0.49	0.30	0.49	0.30	–
Carbon monoxide	8-hour	55	0.12	NA	0.19	0.12	2.35	0.19	2.35	<0.01
	1-hour	NA	0.39	NA	0.58	0.39	7.34	0.58	7.34	<0.01
Sulfur dioxide	Annual	NA	<0.01	NA	<0.01	<0.01	0.04	<0.01	0.04	–
	8-hour	13	<0.01	NA	<0.01	<0.01	0.42	<0.01	0.42	–
	3-hour	NA	<0.01	NA	<0.01	<0.01	1.02	<0.01	1.02	–
Gaseous fluorides	1-month	None	–	NA	–	–	0.14	–	0.14	–
	1-week	NA	–	NA	–	–	0.26	–	0.26	–
	24-hour	NA	–	NA	–	–	0.78	–	0.78	–
	12-hour	NA	–	NA	–	–	1.14	–	1.14	–
Ozone (as VOC)	1-hour	0.2	NC	NA	NC	NC	NC	NC	NC	–

NA = Technology is not applicable to this fuel type.

– = No air emission associated with this option.

NC = Not Calculated.

VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. NIOSH (1991) and OSHA TWAs.

c. Technology options: 1 = Prepare for Direct Disposal/Direct Co-Disposal; 2 = Repackage and Prepare to Ship; 3 = Melt and Dilute; 4 = Mechanical Dilution; 5 = Vittrification Technologies; 6 = Electrometallurgical Treatment; 7 = Conventional Processing; and 8 = Continued Wet Storage.

Table F-3. Estimated maximum incremental concentrations of nonradiological air pollutants for noninvolved worker - HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging (Fuel Group C).^a

			Incremental concentration for technology option ^c							
Pollutant	Averaging time	Regulatory standard ^b	1	2	3	4	5	6	7	8
Toxic pollutants (mg/m ³)										
Nitric acid	24-hour	5	<0.01	NA	<0.01	<0.01	1.30	<0.01	1.30	<0.01
1,1,1-trichloroethane	24-hour	1,900	–	NA	–	–	<0.01	–	<0.01	–
Benzene	24-hour	3.19	–	NA	–	–	<0.01	–	<0.01	–
Ethanolamine	24-hour	6	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	435	–	NA	–	–	<0.01	–	<0.01	–
Ethylene glycol	24-hour	None	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	0.75	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	80	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	0.2	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	1,800	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Manganese	24-hour	5	–	NA	–	–	<0.01	–	<0.01	–
Mercury	24-hour	0.1	–	NA	–	–	–	–	<0.01	–
Methyl alcohol	24-hour	260	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	590	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	410	–	NA	–	–	<0.01	–	<0.01	–
Methylene chloride	24-hour	86.7	–	NA	–	–	<0.01	–	<0.01	–
Napthalene	24-hour	50	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	19	–	NA	–	–	<0.01	–	<0.01	–
Phosphorus	24-hour	0.1	–	NA	–	–	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	2.0	–	NA	–	–	<0.01	–	<0.01	–
Toluene	24-hour	754	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Trichloroethene	24-hour	537	–	NA	–	–	<0.01	–	<0.01	–
Vinyl acetate	24-hour	None	–	NA	–	–	<0.01	–	<0.01	–
Xylene	24-hour	435	<0.01	NA	<0.01	<0.01	0.02	<0.01	0.02	<0.01
Criteria pollutants (µg/m ³)										
Nitrogen oxides	Annual	NA	<0.01	NA	0.01	<0.01	18.17	0.01	18.17	–
Total suspended particulates (total dust)	24-hour	15	<0.01	NA	<0.01	<0.01	0.16	<0.01	0.16	–
Particulate matter (respirable fraction)	Annual	5	0.01	NA	0.01	0.01	<0.01	0.01	<0.01	–
	24-hour	NA	0.16	NA	0.16	0.16	0.10	0.16	0.10	–
Carbon monoxide	8-hour	55	0.04	NA	0.06	0.04	0.78	0.06	0.78	<0.01
	1-hour	NA	0.13	NA	0.19	0.13	2.45	0.19	2.45	<0.01
Sulfur dioxide	Annual	NA	<0.01	NA	<0.01	<0.01	0.01	<0.01	0.01	–
	8-hour	13	<0.01	NA	<0.01	<0.01	0.14	<0.01	0.14	–
	3-hour	NA	<0.01	NA	<0.01	<0.01	0.34	<0.01	0.34	–
Gaseous fluorides	1-month	None	–	NA	–	–	0.05	–	0.05	–
	1-week	NA	–	NA	–	–	0.09	–	0.09	–
	24-hour	NA	–	NA	–	–	0.26	–	0.26	–
	12-hour	NA	–	NA	–	–	0.38	–	0.38	–
Ozone (as VOC)	1-hour	0.2	NC	NA	NC	NC	NC	NC	NC	–

NA = Technology is not applicable to this fuel type.

LEU = low enriched uranium.

— = No air emission associated with this option.

NC = Not Calculated.

HEU = highly enriched uranium.

VOC = volatile organic compound.

- a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.
- b. NIOSH (1991) and OSHA TWAs.
- c. Technology options: 1 = Prepare for Direct Disposal/Direct Co-Disposal; 2 = Repackage and Prepare to Ship; 3 = Melt and Dilute; 4 = Mechanical Dilution; 5 = Vitrification Technologies; 6 = Electrometallurgical Treatment; 7 = Conventional Processing; and 8 = Continued Wet Storage.

Table F-4. Estimated maximum incremental concentrations of nonradiological air pollutants for noninvolved worker - Loose Uranium Oxide in Cans (Fuel Group D).^a

Pollutant	Averaging time	Regulatory standard ^b	Incremental concentration for technology option ^c							
			1	2	3	4	5	6	7	8
Toxic pollutants (mg/m ³)										
Nitric acid	24-hour	5	NA	NA	<0.01	NA	0.13	<0.01	0.13	<0.01
1,1,1-trichloroethane	24-hour	1,900	NA	NA	–	NA	<0.01	–	<0.01	–
Benzene	24-hour	3.19	NA	NA	–	NA	<0.01	–	<0.01	–
Ethanolamine	24-hour	6	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	435	NA	NA	–	NA	<0.01	–	<0.01	–
Ethylene glycol	24-hour	None	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	0.75	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	80	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	0.2	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	1,800	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Manganese	24-hour	5	NA	NA	–	NA	<0.01	–	<0.01	–
Mercury	24-hour	0.1	NA	NA	–	NA	–	–	<0.01	–
Methyl alcohol	24-hour	260	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	590	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	410	NA	NA	–	NA	<0.01	–	<0.01	–
Methylene chloride	24-hour	86.7	NA	NA	–	NA	<0.01	–	<0.01	–
Napthalene	24-hour	50	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	19	NA	NA	–	NA	<0.01	–	<0.01	–
Phosphorus	24-hour	0.1	NA	NA	–	NA	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	2.0	NA	NA	–	NA	<0.01	–	<0.01	–
Toluene	24-hour	754	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Trichloroethene	24-hour	537	NA	NA	–	NA	<0.01	–	<0.01	–
Vinyl acetate	24-hour	None	NA	NA	–	NA	<0.01	–	<0.01	–
Xylene	24-hour	435	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Criteria pollutants (µg/m ³)										
Nitrogen oxides	Annual	NA	NA	NA	<0.01	NA	1.82	<0.01	1.82	–
Total suspended particulates (total dust)	24-hour	15	NA	NA	<0.01	NA	0.02	<0.01	0.02	–
Particulate matter (respirable fraction)	Annual	5	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	–
	24-hour	NA	NA	NA	0.02	NA	0.01	0.02	0.01	–
Carbon monoxide	8-hour	55	NA	NA	<0.01	NA	0.08	<0.01	0.08	<0.01
	1-hour	NA	NA	NA	0.02	NA	0.24	0.02	0.24	<0.01
Sulfur dioxide	Annual	NA	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	–
	8-hour	13	NA	NA	<0.01	NA	0.01	<0.01	0.01	–
	3-hour	NA	NA	NA	<0.01	NA	0.03	<0.01	0.03	–
Gaseous fluorides	1-month	None	NA	NA	–	NA	<0.01	–	<0.01	–
	1-week	NA	NA	NA	–	NA	<0.01	–	<0.01	–
	24-hour	NA	NA	NA	–	NA	0.03	–	0.03	–
	12-hour	NA	NA	NA	–	NA	0.04	–	0.04	–
Ozone (as VOC)	1-hour	0.2	NA	NA	NC	NA	NC	NC	NC	–

NA = Technology is not applicable to this fuel type.

– = No air emission associated with this option.

NC = Not Calculated.

VOC = volatile organic compound.

- Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.
- NIOSH (1991) and OSHA TWAs.
- Technology options: 1 = Prepare for Direct Disposal/Direct Co-Disposal; 2 = Repackage and Prepare to Ship; 3 = Melt and Dilute; 4 = Mechanical Dilution; 5 = Vitrification Technologies; 6 = Electrometallurgical Treatment; 7 = Conventional Processing; and 8 = Continued Wet Storage.

Table F-5. Estimated maximum incremental concentrations of nonradiological air pollutants for noninvolved worker - Higher Actinide Targets (Fuel Group E).^a

Pollutant	Averaging time	Regulatory standard ^b	Incremental concentration for technology option ^c							
			1	2	3	4	5	6	7	8
Toxic pollutants (mg/m³)										
Nitric acid	24-hour	5	NA	–	NA	NA	NA	NA	NA	<0.01
1,1,1-Trichloroethane	24-hour	1,900	NA	–	NA	NA	NA	NA	NA	–
Benzene	24-hour	3.19	NA	–	NA	NA	NA	NA	NA	–
Ethanolamine	24-hour	6	NA	–	NA	NA	NA	NA	NA	<0.01
Ethyl benzene	24-hour	435	NA	–	NA	NA	NA	NA	NA	–
Ethylene glycol	24-hour	None	NA	–	NA	NA	NA	NA	NA	<0.01
Formaldehyde	24-hour	0.75	NA	–	NA	NA	NA	NA	NA	<0.01
Glycol ethers	24-hour	80	NA	–	NA	NA	NA	NA	NA	<0.01
Hexachloronaphthalene	24-hour	0.2	NA	–	NA	NA	NA	NA	NA	<0.01
Hexane	24-hour	1,800	NA	–	NA	NA	NA	NA	NA	<0.01
Manganese	24-hour	5	NA	–	NA	NA	NA	NA	NA	–
Mercury	24-hour	0.1	NA	–	NA	NA	NA	NA	NA	–
Methyl alcohol	24-hour	260	NA	–	NA	NA	NA	NA	NA	<0.01
Methyl ethyl ketone	24-hour	590	NA	–	NA	NA	NA	NA	NA	<0.01
Methyl isobutyl ketone	24-hour	410	NA	–	NA	NA	NA	NA	NA	–
Methylene chloride	24-hour	86.7	NA	–	NA	NA	NA	NA	NA	–
Napthalene	24-hour	50	NA	–	NA	NA	NA	NA	NA	<0.01
Phenol	24-hour	19	NA	–	NA	NA	NA	NA	NA	–
Phosphorus	24-hour	0.1	NA	–	NA	NA	NA	NA	NA	–
Sodium hydroxide	24-hour	2.0	NA	–	NA	NA	NA	NA	NA	–
Toluene	24-hour	754	NA	–	NA	NA	NA	NA	NA	<0.01
Trichloroethene	24-hour	537	NA	–	NA	NA	NA	NA	NA	–
Vinyl acetate	24-hour	None	NA	–	NA	NA	NA	NA	NA	–
Xylene	24-hour	435	NA	–	NA	NA	NA	NA	NA	<0.01
Criteria pollutants (µg/m³)										
Nitrogen oxides	Annual	NA	NA	–	NA	NA	NA	NA	NA	–
Total suspended particulates (total dust)	24-hour	15	NA	–	NA	NA	NA	NA	NA	–
Particulate matter (respirable fraction)	Annual	5	NA	–	NA	NA	NA	NA	NA	–
	24-hour	NA	NA	–	NA	NA	NA	NA	NA	–
Carbon monoxide	8-hour	55	NA	–	NA	NA	NA	NA	NA	<0.01
	1-hour	NA	NA	–	NA	NA	NA	NA	NA	<0.01
Sulfur dioxide	Annual	NA	NA	–	NA	NA	NA	NA	NA	–
	8-hour	13	NA	–	NA	NA	NA	NA	NA	–
	3-hour	NA	NA	–	NA	NA	NA	NA	NA	–
Gaseous fluorides	1-month	None	NA	–	NA	NA	NA	NA	NA	–
	1-week	NA	NA	–	NA	NA	NA	NA	NA	–
	24-hour	NA	NA	–	NA	NA	NA	NA	NA	–
	12-hour	NA	NA	–	NA	NA	NA	NA	NA	–
Ozone (as VOC)	1-hour	0.2	NA	–	NA	NA	NA	NA	NA	–

NA = Technology is not applicable to this fuel type.

– = No air emission associated with this option.

NC = Not Calculated.

VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. NIOSH (1991) and OSHA TWAs.

c. Technology options: 1 = Prepare for Direct Disposal/Direct Co-Disposal; 2 = Repackage and Prepare to Ship; 3 = Melt and Dilute; 4 = Mechanical Dilution; 5 = Vitriification Technologies; 6 = Electrometallurgical Treatment; 7 = Conventional Processing; and 8 = Continued Wet Storage.

Table F-6. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for Uranium and Thorium Metal Fuels (Fuel Group A).^a

Pollutant	Averaging time	Regulatory standard	Incremental concentration for technology option ^b							
			1	2	3	4	5	6	7	8
Toxic pollutants (mg/m ³)										
Nitric acid	24-hour	125	–	NA	–	NA	0.10	–	0.10	–
1,1,1-trichloroethane	24-hour	9,550	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Benzene	24-hour	150	–	NA	–	NA	<0.01	–	<0.01	–
Ethanolamine	24-hour	200	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	–	NA	–	NA	<0.01	–	<0.01	–
Ethylene glycol	24-hour	650	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	15	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	1	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	900	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Manganese	24-hour	25	–	NA	–	NA	<0.01	–	<0.01	–
Mercury	24-hour	0.25	–	NA	–	NA	–	–	<0.01	–
Methyl alcohol	24-hour	1,310	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	–	NA	–	NA	<0.01	–	<0.01	–
Methylene chloride	24-hour	8,750	–	NA	–	NA	<0.01	–	<0.01	–
Napthalene	24-hour	1,250	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	–	NA	–	NA	<0.01	–	<0.01	–
Phosphorus	24-hour	0.5	–	NA	–	NA	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	50	–	NA	–	NA	<0.01	–	<0.01	–
Toluene	24-hour	2,000	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Trichloroethene	24-hour	6,750	–	NA	–	NA	<0.01	–	<0.01	–
Vinyl acetate	24-hour	176	–	NA	–	NA	<0.01	–	<0.01	–
Xylene	24-hour	4,350	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Criteria pollutants (µg/m ³)										
Nitrogen oxides	Annual	100	<0.01	NA	<0.01	NA	1.10	<0.01	1.10	<0.01
Total suspended particulates (total dust)	Annual	75	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Particulate matter (respirable fraction)	Annual	50	–	NA	–	NA	<0.01	–	<0.01	–
Carbon monoxide	24-hour	150	–	NA	–	NA	0.04	–	0.04	–
	8-hour	10,000	0.02	NA	0.03	NA	0.43	0.03	0.43	<0.01
	1-hour	40,000	0.12	NA	0.18	NA	3.20	0.18	3.20	<0.01
Sulfur dioxide	Annual	80	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	–
	8-hour	365	<0.01	NA	0.01	NA	0.04	0.01	0.04	–
	3-hour	1,300	–	NA	–	NA	0.32	–	0.32	–
Gaseous fluorides	1-month	0.8	–	NA	–	NA	<0.01	–	<0.01	–
	1-week	1.6	–	NA	–	NA	0.01	–	0.01	–
	24-hour	2.9	–	NA	–	NA	0.02	–	0.02	–
	12-hour	3.7	–	NA	–	NA	0.04	–	0.04	–
Ozone (as VOC)	1-hour	235	0.05	NA	0.07	NA	0.26	0.08	0.26	–

NA = Technology is not applicable to this fuel type.

– = No air emission associated with this option.

+= No state standard.

VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. Technology options: 1 = Prepare for Direct Disposal/Direct Co-Disposal; 2 = Repackage and Prepare to Ship; 3 = Melt and Dilute; 4 = Mechanical Dilution; 5 = Vitrification Technologies; 6 = Electrometallurgical Treatment; 7 = Conventional Processing; and 8 = Continued Wet Storage.

Table F-7. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for Materials Test Reactor-Like Fuels (Fuel Group B).^a

Pollutant	Averaging time	Regulatory standard	Incremental concentration for technology option ^b							
			1	2	3	4	5	6	7	8
Toxic pollutants (mg/m ³)										
Nitric acid	24-hour	125	–	NA	–	–	0.15	–	0.15	–
1,1,1-trichloroethane	24-hour	9,550	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Benzene	24-hour	150	–	NA	–	–	<0.01	–	<0.01	–
Ethanolamine	24-hour	200	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	–	NA	–	–	<0.01	–	<0.01	–
Ethylene glycol	24-hour	650	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	15	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	1	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	900	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Manganese	24-hour	25	–	NA	–	–	<0.01	–	<0.01	–
Mercury	24-hour	0.25	–	NA	–	–	–	–	<0.01	–
Methyl alcohol	24-hour	1310	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	–	NA	–	–	<0.01	–	<0.01	–
Methylene chloride	24-hour	8,750	–	NA	–	–	<0.01	–	<0.01	–
Napthalene	24-hour	1,250	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	–	NA	–	–	<0.01	–	<0.01	–
Phosphorus	24-hour	0.5	–	NA	–	–	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	50	–	NA	–	–	<0.01	–	<0.01	–
Toluene	24-hour	2,000	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Trichloroethene	24-hour	6,750	–	NA	–	–	<0.01	–	<0.01	–
Vinyl acetate	24-hour	176	–	NA	–	–	<0.01	–	<0.01	–
Xylene	24-hour	4,350	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Criteria pollutants (µg/m ³)										
Nitrogen oxide	Annual	100	<0.01	NA	<0.01	<0.01	1.65	<0.01	1.65	<0.01
Total suspended particulates (total dust)	Annual	75	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Particulate matter (respirable fraction)	Annual	50	–	NA	–	–	<0.01	–	<0.01	–
	24-hour	150	–	NA	–	–	0.06	–	0.06	–
Carbon monoxide	8-hour	10,000	0.03	NA	0.05	0.03	0.65	0.05	0.65	<0.01
	1-hour	40,000	0.18	NA	0.27	0.18	4.80	0.27	4.80	<0.01
Sulfur dioxide	Annual	80	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	–
	8-hour	365	0.01	NA	0.02	0.01	0.06	0.02	0.06	–
	3-hour	1,300	–	NA	–	–	0.48	–	0.48	–
Gaseous fluorides	1-month	0.8	–	NA	–	–	<0.01	–	<0.01	–
	1-week	1.6	–	NA	–	–	0.02	–	0.02	–
	24-hour	2.9	–	NA	–	–	0.03	–	0.03	–
	12-hour	3.7	–	NA	–	–	0.06	–	0.06	–
Ozone (as VOC)	1-hour	235	0.08	NA	0.11	0.08	0.39	0.11	0.39	–

NA = Technology is not applicable to this fuel type.

– = No air emission associated with this option.

+ = No state standard.

VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. Technology options: 1 = Prepare for Direct Disposal/Direct Co-Disposal; 2 = Repackage and Prepare to Ship; 3 = Melt and Dilute; 4 = Mechanical Dilution; 5 = Vitrification Technologies; 6 = Electrometallurgical Treatment; 7 = Conventional Processing; and 8 = Continued Wet Storage.

Table F-8. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging (Fuel Group C).^a

Pollutant	Averaging time	Regulatory standard	Incremental concentration for technology option ^b							
			1	2	3	4	5	6	7	8
Toxic pollutants (mg/m ³)										
Nitric acid	24-hour	125	–	NA	–	–	0.05	–	0.05	–
1,1,1-trichloroethane	24-hour	9,550	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Benzene	24-hour	150	–	NA	–	–	<0.01	–	<0.01	–
Ethanolamine	24-hour	200	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	–	NA	–	–	<0.01	–	<0.01	–
Ethylene glycol	24-hour	650	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	15	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	1	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	900	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Manganese	24-hour	25	–	NA	–	–	<0.01	–	<0.01	–
Mercury	24-hour	0.25	–	NA	–	–	–	–	<0.01	–
Methyl alcohol	24-hour	1310	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	–	NA	–	–	<0.01	–	<0.01	–
Methylene chloride	24-hour	8,750	–	NA	–	–	<0.01	–	<0.01	–
Napthalene	24-hour	1,250	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	–	NA	–	–	<0.01	–	<0.01	–
Phosphorus	24-hour	0.5	–	NA	–	–	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	20	–	NA	–	–	<0.01	–	<0.01	–
Toluene	24-hour	2,000	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Trichloroethene	24-hour	6,750	–	NA	–	–	<0.01	–	<0.01	–
Vinyl acetate	24-hour	176	–	NA	–	–	<0.01	–	<0.01	–
Xylene	24-hour	4,350	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Criteria pollutants (µg/m ³)										
Nitrogen oxide	Annual	100	<0.01	NA	<0.01	<0.01	0.55	<0.01	0.55	<0.01
Total suspended particulates (total dust)	Annual	75	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Particulate matter (respirable fraction)	Annual	50	–	NA	–	–	<0.01	–	<0.01	–
	24-hour	150	–	NA	–	–	0.02	–	0.02	–
Carbon monoxide	8-hour	10,000	0.01	NA	0.02	0.01	0.22	0.02	0.22	<0.01
	1-hour	40,000	0.06	NA	0.09	0.06	1.60	0.09	1.60	<0.01
Sulfur dioxide	Annual	80	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	–
	8-hour	365	<0.01	NA	<0.01	<0.01	0.02	<0.01	0.02	–
	3-hour	1,300	–	NA	–	–	0.16	–	0.16	–
Gaseous fluorides	1-month	0.8	–	NA	–	–	<0.01	–	<0.01	–
	1-week	1.6	–	NA	–	–	<0.01	–	<0.01	–
	24-hour	2.9	–	NA	–	–	0.01	–	0.01	–
	12-hour	3.7	–	NA	–	–	0.02	–	0.02	–
Ozone (as VOC)	1-hour	235	0.03	NA	0.04	0.03	0.13	0.04	0.13	–

NA = Technology is not applicable to this fuel type.

– = No air emission associated with this option.

+= No state standard.

HEU = highly enriched uranium.

LEU = low enriched uranium.

VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. Technology options: 1 = Prepare for Direct Disposal/Direct Co-Disposal; 2 = Repackage and Prepare to Ship; 3 = Melt and Dilute; 4 = Mechanical Dilution; 5 = Vitrification Technologies; 6 = Electrometallurgical Treatment; 7 = Conventional Processing; and 8 = Continued Wet Storage.

Table F-9. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for Loose Uranium Oxide in Cans (Fuel Group D).^a

Pollutant	Averaging time	Regulatory standard	Incremental concentration for technology option ^b							
			1	2	3	4	5	6	7	8
Toxic pollutants (mg/m ³)										
Nitric acid	24-hour	125	NA	NA	–	NA	<0.01	–	<0.01	–
1,1,1-trichloroethane	24-hour	9,550	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Benzene	24-hour	150	NA	NA	–	NA	<0.01	–	<0.01	–
Ethanolamine	24-hour	200	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	NA	NA	–	NA	<0.01	–	<0.01	–
Ethylene glycol	24-hour	650	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	7.5	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	1	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	200	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Manganese	24-hour	25	NA	NA	–	NA	<0.01	–	<0.01	–
Mercury	24-hour	0.25	NA	NA	–	NA	–	–	<0.01	–
Methyl alcohol	24-hour	1,310	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	NA	NA	–	NA	<0.01	–	<0.01	–
Methylene chloride	24-hour	8,750	NA	NA	–	NA	<0.01	–	<0.01	–
Napthalene	24-hour	1,250	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	NA	NA	–	NA	<0.01	–	<0.01	–
Phosphorus	24-hour	0.5	NA	NA	–	NA	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	20	NA	NA	–	NA	<0.01	–	<0.01	–
Toluene	24-hour	2,000	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Trichloroethene	24-hour	6,750	NA	NA	–	NA	<0.01	–	<0.01	–
Vinyl acetate	24-hour	176	NA	NA	–	NA	<0.01	–	<0.01	–
Xylene	24-hour	4,350	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Criteria pollutants (µg/m ³)										
Nitrogen oxide	Annual	100	NA	NA	<0.01	NA	0.06	<0.01	0.06	<0.01
Total suspended particulates (total dust)	Annual	75	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Particulate matter (respirable fraction)	Annual	50	NA	NA	–	NA	<0.01	–	<0.01	–
	24-hour	150	NA	NA	–	NA	<0.01	–	<0.01	–
Carbon monoxide	8-hour	10,000	NA	NA	<0.01	NA	0.02	<0.01	0.02	<0.01
	1-hour	40,000	NA	NA	<0.01	NA	0.16	<0.01	0.16	<0.01
Sulfur dioxide	Annual	80	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	–
	8-hour	365	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	–
	3-hour	1,300	NA	NA	–	NA	0.02	–	0.02	–
Gaseous fluorides	1-month	0.8	NA	NA	–	NA	<0.01	–	<0.01	–
	1-week	1.6	NA	NA	–	NA	<0.01	–	<0.01	–
	24-hour	2.9	NA	NA	–	NA	<0.01	–	<0.01	–
	12-hour	3.7	NA	NA	–	NA	<0.01	–	<0.01	–
Ozone (as VOC)	1-hour	235	NA	NA	<0.01	NA	0.01	<0.01	0.01	–

NA = Technology is not applicable to this fuel type.

– = No air emission associated with this option.

+ = No state standard.

VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. Technology options: 1 = Prepare for Direct Disposal/Direct Co-Disposal; 2 = Repackage and Prepare to Ship; 3 = Melt and Dilute; 4 = Mechanical Dilution; 5 = Vittrification Technologies; 6 = Electrometallurgical Treatment; 7 = Conventional Processing; and 8 = Continued Wet Storage.

Table F-10. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for Higher Actinide Targets (Fuel Group E).^a

Boundary for Higher Pollutant Targets (Fast Group B)			Incremental concentration for technology option ^b							
Pollutant	Averaging time	Regulatory standard	1	2	3	4	5	6	7	8
Toxic pollutants (mg/m³)										
Nitric acid	24-hour	125	NA	–	NA	NA	NA	NA	NA	–
1,1,1-trichloroethane	24-hour	9,550	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Benzene	24-hour	150	NA	-	NA	NA	NA	NA	NA	–
Ethanolamine	24-hour	200	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Ethyl benzene	24-hour	4,350	NA	-	NA	NA	NA	NA	NA	-
Ethylene glycol	24-hour	650	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Formaldehyde	24-hour	15	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Glycol ethers	24-hour	+	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Hexachloronaphthalene	24-hour	1	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Hexane	24-hour	900	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Manganese	24-hour	25	NA	–	NA	NA	NA	NA	NA	–
Mercury	24-hour	0.25	NA	–	NA	NA	NA	NA	NA	–
Methyl alcohol	24-hour	1,310	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Methyl ethyl ketone	24-hour	14,750	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Methyl isobutyl ketone	24-hour	2,050	NA	–	NA	NA	NA	NA	NA	–
Methylene chloride	24-hour	515	NA	–	NA	NA	NA	NA	NA	–
Napthalene	24-hour	1,250	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Phenol	24-hour	190	NA	–	NA	NA	NA	NA	NA	–
Phosphorus	24-hour	0.5	NA	–	NA	NA	NA	NA	NA	–
Sodium hydroxide	24-hour	50	NA	–	NA	NA	NA	NA	NA	–
Toluene	24-hour	2,000	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Trichloroethene	24-hour	6,750	NA	–	NA	NA	NA	NA	NA	–
Vinyl acetate	24-hour	176	NA	–	NA	NA	NA	NA	NA	–
Xylene	24-hour	4,350	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Criteria pollutants (µg/m³)										
Nitrogen oxide	Annual	100	NA	–	NA	NA	NA	NA	NA	<0.01
Total suspended particulates (total dust)	Annual	75	NA	–	NA	NA	NA	NA	NA	<0.01
Particulate matter (respirable fraction)	Annual	50	NA	–	NA	NA	NA	NA	NA	–
	24-hour	150	NA	–	NA	NA	NA	NA	NA	–
Carbon monoxide	8-hour	10,000	NA	–	NA	NA	NA	NA	NA	<0.01
	1-hour	40,000	NA	–	NA	NA	NA	NA	NA	<0.01
Sulfur dioxide	Annual	80	NA	–	NA	NA	NA	NA	NA	–
	8-hour	365	NA	–	NA	NA	NA	NA	NA	–
	3-hour	1,300	NA	–	NA	NA	NA	NA	NA	–
Gaseous fluorides	1-month	0.8	NA	–	NA	NA	NA	NA	NA	–
	1-week	1.6	NA	–	NA	NA	NA	NA	NA	–
	24-hour	2.9	NA	–	NA	NA	NA	NA	NA	–
	12-hour	3.7	NA	–	NA	NA	NA	NA	NA	–
Ozone (as VOC)	1-hour	245	NA	<0.01	NA	NA	NA	NA	NA	–

NA = Technology is not applicable to this fuel type.

– = No air emission associated with this option.

+= No state standard.

VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. Technology options: 1 = Prepare for Direct Disposal/Direct Co-Disposal; 2 = Repackage and Prepare to Ship; 3 = Melt and Dilute; 4 = Mechanical Dilution; 5 = Vitrification Technologies; 6 = Electrometallurgical Treatment; 7 = Conventional Processing; and 8 = Continued Wet Storage.

APPENDIX G. PUBLIC COMMENTS AND DOE RESPONSES

This section provides DOE's responses to comments received during the public comment period. Comments received during the public meetings in Columbia and North Augusta, South Carolina are summarized. Letters received are reproduced in this section. The transcripts from the meetings can be reviewed at the DOE public reading rooms: DOE Freedom of Information Reading Room, Forrestal Building, Room 1E-190, 1000 Independence Ave., S.W., Washington D.C., 20585, Phone: 202-586-6020 and DOE Public Document Room, University of South Carolina, Aiken Campus, University Library, 2nd Floor, 171 University Parkway, Aiken, S.C. 29801, Phone: 803-648-6851.

DOE published the *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement* (DOE/EIS-0279D) in December 1998. DOE held public meetings on the Draft Environmental Impact Statement (DEIS) in Columbia, South Carolina on January 28, 1999 and in North Augusta, South Carolina on February 2, 1999. The public comment period ended on February 8, 1999.

Court reporters documented comments and statements made during four public meeting sessions. In those sessions, 17 individuals provided comments or made public statements. DOE also received 15 letters on the Draft EIS by U.S. mail or by electronic mail.

EC | DOE also received seven letters commenting on the EIS after February 8, 1999, and the comments have been addressed in the final EIS.

This appendix presents the comments received and the DOE responses to those comments. If a comment prompted a modification to the EIS, DOE has noted the change and directed the reader to that change.

EC | For ease in summarizing the comments, DOE divided the comments into 12 major categories. For each of these categories, DOE identified the number of comments received and provided a general summary of comments. The major points and DOE's responses are summarized below:

Processing

EC | Comments were received related to processing of SNF. These ranged from support of proc-

essing as a proven method for disposition of SNF to admonitions that the processing facilities (canyons) should be shut down immediately. A number of comments asked for clarification regarding the criteria used when determining when processing would be necessary for SNF currently in storage. Commentors also criticized the method by which DOE outlined the missions of the canyons and several requested definite closure dates for the canyons.

| EC

| EC

Response: The canyons at SRS cannot be shut down immediately because DOE is utilizing these facilities to stabilize nuclear materials. In this EIS, DOE proposes to use the canyons to process a relatively small amount (about 3 percent by volume or 40% by weight) of the SNF under consideration to eliminate the potential for certain health and safety problems. The basis for selecting the SNF proposed for processing is discussed in Section 2.4.3.2. DOE estimates the processing time would be less than 6 months in F Canyon and about 1 year in H Canyon. The proposed processing operations are within the current canyon schedule planning basis. In other words, the proposed SNF processing activities would not extend the planned canyon operations. However, establishing closure dates for the SRS canyons is beyond scope of this EIS.

| TC

| EC

Alternatives

Comments were received regarding alternatives to conventional processing of SNF. The comments ranged from support for alternatives to conventional processing to concerns about technical adequacy. Commentors also questioned

| EC

| EC

EC | DOE's ability to develop new technology to treat SNF in a timely manner.

Response: DOE evaluated a variety of technologies for managing aluminum-based SNF at the SRS. DOE considers that the range of technologies included in the EIS to be an appropriate reflection of the technologies available. DOE also considers the range of alternatives evaluated in the EIS to represent a reasonable set of the technology combinations that could have been evaluated. Public comment did not reveal an SNF management technology that DOE has not considered. The DOE has completed considerable research and development work for the proposed SNF alternative treatment technology (i.e., Melt and Dilute). As part of the Department's Fiscal Year 2000 budget, DOE requested funds from Congress for research and development work, and funds to begin facility design. DOE is committed to developing and implementing that technology for aluminum-based SNF.

Waste Form

EC | Comments were received relating to the acceptability in any geologic repository of the SNF waste form that would be produced using a new (alternative) SNF treatment technology. The principal concern was that waste acceptance criteria for a geologic repository have not been established. In this regard, the concern was that alternative technologies for the disposition of SNF may produce a product that will not meet the final repository criteria.

EC | Response: Waste acceptance criteria describe the physical, chemical, and thermal characteristics to which SNF and associated canisters must conform for emplacement in a geologic repository. DOE has assessed the waste forms the primary new SNF treatment technologies (Melt and Dilute, and Direct Disposal/Direct Co-Disposal) would generate against repository preliminary waste acceptance criteria. DOE concluded that waste forms from both technologies would meet the preliminary criteria. Section 2.2.1 has been revised to discuss the issue in greater detail.

Impacts

Comments were received related to the calculation of impacts from the proposed actions. These comments ranged from expressions that specific impacts were negligible to comments that the impacts from past site activities had been underestimated.

Response: The impact estimates in the Final EIS are based on data from site operations or operating experience and the judgement of expert analysts. DOE believes the analyses presented in the EIS are adequate for comparing SNF management alternatives.

Openness/Independent Review

Comments were received regarding independent reviews of the SNF treatment technologies and how they would be used in the decisionmaking process. The comments called for increased public involvement. Some comments also called for DOE to share technology development data, particularly regarding the requirements and performance of the off-gas system for the proposed Melt and Dilute process.

Response: DOE believes that the EIS process provides adequate opportunities for public involvement. For example, DOE has invited public comment and input for this process during scoping meetings and during the public comment period for the Draft EIS. Information regarding technology development, that is referenced in the Final EIS, is available at the DOE Public Reading Room, University of South Carolina at Aiken, Gregg-Graniteville Library, University Parkway, Aiken, South Carolina and at the DOE Freedom of Information Reading Room (Room 1E-190), Forrestal Building, 1000 Independence Ave., Washington, D.C. Additionally, information regarding the development of the new SNF treatment technologies, including the off-gas system that would be used to collect fission products that could evolve during operation of the Melt and Dilute process, may be requested from Randall Ponik, U.S. Department of Energy, P.O. Box A, Aiken, South Carolina 29802, (803) 952-2549 or via E-mail at

randall.ponik@SRS.gov. DOE has also solicited advice, comments, and help with development on the Melt and Dilute process from outside the Department. Contributing institutions include the University of South Carolina, the U.S. Nuclear Regulatory Commission (NRC), and the National Academy of Sciences. Their reports are publicly available, including in the DOE reading rooms, or upon request.

Cost

Comments were received regarding the potential costs of SNF treatment alternatives. These comments primarily questioned whether all costs and credits had been considered. This included the credits for the separation and sale of usable enriched uranium to the commercial nuclear power industry. Comments also were made regarding the adequacy of funding for the implementation of SNF treatment alternatives.

Response: DOE prepared a report on costs associated with aluminum-based SNF treatment technologies. DOE considered all appropriate factors to prepare the report. A discussion of uranium credits (i.e., cost recovery based on the sale of low-enriched uranium to the private sector) was included in the cost analysis. The results of the cost report are discussed in Section 2.6.5. DOE obtained an independent review of the cost report; the recommendations from the independent review were factored into the report.

References

Comments were received related to the references used in the preparation of the EIS. The comments generally suggested alternate sources of information for the EIS and suggested that both foreign and domestic SNF handling experience be included in the discussion.

Response: DOE considered the suggestions. Based on reports cited by the commentor, DOE believes the most accurate and current information was used to prepare the Final EIS. The information is based on actual Site operations (e.g., handling foreign and domestic SNF) and

conditions or on estimates of operational activities and conditions that would exist for new facilities. As a result, DOE believes that data and references used in preparing the EIS provide an adequate basis for estimating impacts and for comparing technologies and alternatives. Current regulatory requirements and information have been cited as applicable.

Nonproliferation

Comments were received regarding nonproliferation issues as they relate to the treatment and disposition of SNF. A number of commenters felt that nonproliferation was being overemphasized in relation to its importance. However, one commentor doubted the independence of DOE in the preparation of the nonproliferation study and another commentor stated that DOE should take a worldwide leadership role in nonproliferation by treating SNF without separating potential weapons materials.

Response: DOE believes that nonproliferation to be one of the factors that should be considered during the decision-making process. DOE conducted a nonproliferation study for SNF treatment technologies in conjunction with the preparation of the Draft EIS. The study concluded that all technologies considered in the Draft EIS were compatible with the nonproliferation goals of the United States but that separations technologies, such as Conventional Processing, had distinct disadvantages because fissile material would be separated. The study was reviewed by a panel of experts independent of DOE: Matthew Bunn, Belfor Center for Science and International Affairs, Harvard University; Frank von Hippel, Woodrow Wilson School of Public and International Affairs, Princeton University; George Bunn, Center for International Security and Cooperation, Institute for International Studies, Stanford University; Harold Bengelsdorf, Bengelsdorf, McGoldrick and Associates, LLC; and David Albright, Institute for Science and International Security. No problems were identified with the conclusions presented in the report.

Methodology

Comments were received related to the methodologies used in the preparation of the EIS. These included both positive and negative comments on health issues and environmental justice. One of the commenters asked what environmental impact would result from the release of cesium into the atmosphere in the event that the filtration system does not capture all the cesium. Another commenter stated that DOE had minimized impacts in the Cumulative Impacts Chapter and only used a limited amount of available information regarding actual operating experience. The Environmental Protection Agency commended DOE on their method of segregating spent fuel by type and then applying the appropriate treatment methodology as the best way to proceed.

EC

Response: Impacts in the EIS are estimated from the best available information, including operational data whenever possible. When operations data do not exist, SRS uses experience and information from similar facilities and the best judgement of technical experts. DOE believes that the methodology used in the EIS is adequate for comparing SNF management alternatives.

EC

Purpose and Need

Comments were received related to the stated purpose and need for agency action. The comments generally focused on long-term solutions to the problems of SNF poses and noted that other nuclear materials that could be processed in SRS facilities should also be addressed.

EC

Response: DOE proposes to implement a technology that will prepare aluminum-based SNF to meet the requirements for disposal in a geologic repository, and will make the SNF ready for offsite shipment. DOE is separately evaluating potential geologic disposal of high-level waste and spent nuclear fuel in the Yucca Mountain Repository EIS, as discussed in Section 1.6 of the SNF Manage-

EC

ment EIS. DOE has addressed other nuclear materials that could be managed at the SRS as part of the cumulative impacts discussion in Chapter 5 of this EIS. The inventory of material was based on recent studies completed by DOE (see Chapter 5).

Safety

Comments were received related to general safety issues of the proposed actions. Most comments were related to concerns that facilities would be constructed and operated using stringent safety standards.

Response: DOE is committed to the protection of workers, the public and the environment. All operations and facilities at SRS meet or exceed all applicable health protection and safety requirements. SNF treatment facilities and operations will meet or exceed all applicable requirements.

Failure of Stored SNF Before Treatment

Comments were received regarding the possibility that stored SNF could fail before proposed treatment facilities are available. The comments requested impact estimates for these potential failures.

EC

Response: The preferred alternative in the Final EIS includes a discussion of the action that would be taken (processing in SRS canyons) should SNF fail while in storage pending implementation of a new treatment technology. Section 1.5 includes a qualitative discussion of the types of health and safety issues (e.g., uncontrolled release of fission products into storage basin water) that would be created by the failure of the SNF that DOE believes presents certain vulnerabilities for continued storage.

In addition, a number of other comments were received that offered editorial suggestions, could not be easily categorized, or were deemed to be out of scope of this EIS.

Comments are identified by one of the following letter codes:

- L1- through L15- (comments received by letter or email)
- M1- or M2- (comments submitted in the afternoon and evening session of the public meetings in Columbia, SC)
- M4- (comments received in the evening session of the public meetings in North Augusta, SC. The afternoon session in North Augusta did not generate any comments)

DOE numbered specific comments in each letter and meeting sequentially (01, 02, etc.) to pro-

vide unique identifiers. Table G-1 lists the individuals and government agencies that submitted comments and their unique identifiers.

The Department extends its gratitude to all the individuals and agencies who showed the interest and took the time to provide comments.

LETTERS

The comment letters DOE received on the Draft SNF Management EIS and DOE's responses are provided in the following section. Comments in each letter are identified, and the corresponding responses follow the letter.

Table G-1. Public comments on the Draft SNF EIS.

Comment source number	Commenter	Page number
L1	U.S. Environmental Protection Agency	G-7 to G-8
L2	Institute for Energy and Environmental Research	G-10 to G-13
L3	Environmentalists, Inc.	G-19 to G-21
L4	Nuclear Control Institute	G-25 to G-28
L5	Economic Development Partnership	G-31 to G-35
L6	SRS Citizens Advisory Board	G-38 to G-39
L7	Mr. Robert F. Overman	G-41
L8	Ms. Christine Witkowski	G-43 to G-44
L9	Ms. Christine Witkowski	G-47
L10	U.S. Department of Health and Human Services	G-49
L11	National Oceanic and Atmospheric Administration	G-51 to G-52
L12	Mr. Stan and Ms. Elaine Frick	G-54
L13	Mr. William A. Lochstet	G-56 to G-58
L14	Mr. Charles R. Goergen	G-60 to G-62
L15	Environmentalists Inc.	G-64 to G-65
L16	Nuclear Control Institute/Natural Resources Defense Council/Institute for Science and International Technology/Dr. F. von Hippel	G-67 to 70
L17	Defense Nuclear Facilities Safety Board	G-74 to 119
L18	Citizens for Nuclear Technology Awareness	G-130 to 133
L19	State of South Carolina State Budget and Control Board	G-137 to 143
L20	South Carolina Department of Natural Resources	G-145 to 146
L21	National Oceanic and Atmospheric Administration	G-148 to 150

Comment source number ^a	Commenter
M1-01, 03 – 05, 07, 08, 13, 14, 18 – 21, 44 – 45	Mr. Harry Rogers
M1-02	Ms. Meira Warshauer
M1-06, 43	Dr. Linda Samuel
M1-09	Dr. Mary Kelly
M1-10, 17, 33 – 42	Mr. Robert Guild
M1-11, 12	Mr. Claude Gilbert
M1-15	Ms. Ann Clark
M1-16, 22	Mr. David Wilson
M1-23 – 32	Mr. Ernie Chaput
M2-01 – 04, 23 – 26	Mr. Brett Bursey
M2-05, 06, 09, 13 – 14, 18 – 22, 27, 28, 29	Ms. Christine Witkowski
M2-07 – 08, 16 – 17	Mr. Robert Witkowski
M2-10 – 12	Ms. Susan Corbett
M2-15	Mr. Rex Henry
M4-01, 18	Mr. Chuck Goergen
M4-02, 03, 10, 11, 17	Mr. Rick Geddes
M4-04 – 09, 12 – 16	Mr. Lee Poe

a. Unique source codes were given to each of the public meeting sessions (M1 – M4, respectively). The individual's comments are coded M1-01, etc.

b. Complete transcripts of the meetings are available in the DOE Public Reading Rooms.



UNITED STATES ENVIRONMENTAL PROTECTION AGENCY
REGION 4
ATLANTA FEDERAL CENTER
61 FORSYTH STREET
ATLANTA, GEORGIA 30303-8960

February 8, 1999

4EAD/OEA

Mr. Andrew R. Grainger
NEPA Compliance Officer
Savannah River Site
Building 742-A, Room 185
Aiken, SC 29802

SUBJECT: *Savannah River Site Spent Nuclear Fuel Management*
Draft Environmental Impact Statement (DOE/EIS-0279D)
Aiken, South Carolina

Dear Mr. Grainger:

We have reviewed the subject document in accordance with Section 102(2)(C) of the National Environmental Policy Act (NEPA) and Section 309 of the Clean Air Act. The DEIS discusses alternatives for management of spent nuclear fuel (SNF) at the Savannah River Site (SRS). The DOE anticipates the need for storage of these SNFs until their ultimate disposition in an off-site geologic repository to be constructed.

The DOE identifies the melt and dilute treatment option as the preferred method of managing most of the aluminum-based SNF. Conventional chemical processing is also identified as the preferred method for the remaining 3% by volume of the spent fuels. The preferred alternative includes a new packaging technology option, new processing technology option, and the conventional processing technology option.

Overall, the DEIS does a good job dealing with complex issues. The summary document was concise and helpful. While our review identified no major technical deficiencies, we offer the attached comments and observations.

Based on our comments, we rate this DEIS "EC-2." That is, we have environmental concerns about the project and more information is needed to fully assess the impacts. If you have any questions concerning our comments, you may contact Ramona McConney of my staff at 404/562-9615.

Sincerely,

A handwritten signature in cursive script, reading "Heinz J. Mueller", is written over the typed name.

Heinz J. Mueller, Chief
Office of Environmental Assessment

Attachment

Comments on
Savannah River Site Spent Nuclear Fuel Management
Draft Environmental Impact Statement (DOE/EIS-0279D)

ENVIRONMENTAL JUSTICE

We commend DOE on their assessment of environmental justice (Section 4.1.1.6). The DEIS concludes that none of the alternative strategies would have disproportionate adverse effects on minority populations or low-income communities (page 4-30).

L1-1

POLLUTION PREVENTION

In the interests of pollution prevention, the alternative(s) that involve the least transportation and least new construction should be seriously considered. EPA concurs with DOE's preference for brownfields redevelopment if construction of a transfer/storage/treatment facility is necessary (page 4-48).

L1-2

SUMMARY

We agree that the No Action Alternative, (continued long-term underwater storage of aluminum-based SNF), is not a preferable option, since it could lead to increased corrosion with increased environmental health and safety vulnerabilities (page 4-37).

L1-3

Segregating spent fuels by type with the goal of applying appropriate treatment methodologies to a given fuel is the best way to proceed. Perhaps the most critical technical problem is how highly enriched uranium should be treated prior to preparing it for storage in a repository. The melt-dilute process places U-235 in a configuration where the potential for a critical accident is reduced. The other technologies identified in this document can be applied as appropriate to fuel groups that don't present criticality concerns.

L1-4

For any alternative chosen, EPA emphasizes that pollution prevention and waste minimization should be considered processes of continuous improvement to be integrated into every waste management activity.

L1-5

Response to comment L1-1: Comment noted.

Response to comment L1-2: The Department of Energy fully supports pollution prevention programs and has taken pollution prevention into consideration in the evaluation of alternatives for management of spent nuclear fuel at the Savannah River Site. The alternative that involves the least transportation and the least construction (other than the No-Action Alternative) is the Maximum Impact Alternative, which uses Conventional Processing to produce 160 DWPF canisters that would be shipped to a geologic repository. The Preferred Alternative would produce about 400 SNF canisters that would be shipped to a geologic repository. The other alternatives would produce up to 1,400 canisters. The number of SNF or DWPF canisters that would be produced under each alternative was one of the factors DOE considered in identifying its preferred alternative for SNF management.

As discussed in the Summary of the EIS, to implement the preferred alternative, DOE would construct a melt and dilute facility in the existing 105-L building at SRS and build a dry-storage facility in L-Area, near the 105-L building. Reuse of an existing facility, rather than new construction, is consistent with DOE's commitment to Pollution Prevention.

Response to comment L1-3: Comment noted.

Response to comment L1-4: Comment noted. As described in Appendix D of the EIS, the frequency of a criticality in the wet basins ranges from once in 300 years (Reactor Disassembly Basin) to once in 1,500 years (Receiving Basin for Offsite Fuel). The frequency of a criticality in the dry storage facility (including storage of melt and dilute products) is once in 33,000 years.

In addition, the melt and dilute product form is being evaluated by the NRC as part of the Waste Acceptance Criteria review for the proposed geologic repository. The Waste Acceptance Criteria for the geologic repository would have criticality controls stipulated. This is why depleted uranium is being used to bring the melt

and dilute product down to low enriched uranium and why neutron poisons may be used as well. Both of these factors will allow final storage of the melt and dilute form to be acceptable from a criticality standpoint.

Response to comment L1-5: Pollution prevention and waste minimization are, and will continue to be, integral parts of SNF management at SRS. DOE has implemented an aggressive waste minimization and pollution prevention program at SRS. As a result, significant reductions have been achieved in the amounts of wastes discharged into the environment or sent to landfills, and significant cost savings have been realized from re-cycling or selling usable materials. Consistent with this program, the construction and operation of SNF management facilities would be evaluated to characterize waste streams and identify alternatives for reducing or eliminating them. Emphasis would be placed on minimizing low-level radioactive waste, the largest waste stream through source reduction and recycling. Selected waste minimization practices could include the following:

- Changes to process design to reduce the potential for spills
- Decontaminate equipment to facilitate recycling or reuse
- Recycle metals and other usable materials
- Preventive maintenance to extend process equipment life
- Utilize modular equipment designs to isolate potential failure elements to avoid changing out entire units

During the construction phase of the project, emphasis would be placed on reutilization of construction materials to minimize waste to the landfills.

Section 4.1.1.4 and Section 6.5.1 have been revised to discuss waste minimization programs.

TC

EC

TC

**INSTITUTE FOR ENERGY AND
ENVIRONMENTAL RESEARCH**

February 8, 1999

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Mr. Drew Grainger
NEPA Compliance Officer
U.S. Department of Energy
Savannah River Operations Office
Building 742A, Room 183
Aiken, SC 29802

Re: Savannah River Site (SRS) Spent Nuclear Fuel (SNF) Management Draft Environmental
Impact Statement (EIS), DOE/EIS-0279D

Dear Mr. Grainger,

Following are comments of the Institute for Energy and Environmental Research regarding the above referenced Draft EIS. We support the Department of Energy's (DOE) efforts, as expressed in the preferred alternative, to deploy techniques other than conventional reprocessing. Demonstrating that nuclear wastes, such as spent fuel, can be managed without separation is essential to sound nonproliferation policy, and ending reprocessing at SRS is a critical step in reducing risks and saving money at that facility.

L2-1

Still, we are concerned about shortcomings in the Draft EIS, especially in two areas. First, the discussion of reprocessing does not provide a convincing rationale for the proposed use of the canyons. Second, the Draft EIS does not lay out a clear plan for ensuring that the technical uncertainties associated with the melt and dilute option will be resolved so that the technology can be implemented within the next five years.

L2-2

L2-3

Reprocessing is not DOE's preferred alternative for most aluminum-based SNF, but the Draft EIS does open the door wide to the possibility that DOE will reprocess at least some of this fuel. The Draft EIS states DOE could reprocess aluminum-based SNF if "any health or safety concerns" are identified prior to the melt and dilute facility becoming operational. (p. S-18) This is a meaningless criteria for future decision making. One could make a sound case that there is an inherent health and safety concern merely because the SNF exists and is, or will be, in storage at SRS. The Final EIS should lay out meaningful criteria by which DOE will determine if the health and safety concern is sufficient to require urgent action and also consider interim measures which could be taken to minimize risks until a melt and dilute facility is operational.

L2-4

DOE has identified reprocessing as the preferred alternative for several types of material. Fuels from the Experimental Breeder Reactor-II (EBR-II) and the Sodium Reactor Experiment are



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proposed for reprocessing because they have been declad and so would oxidize rapidly if exposed to water. The result would be a release of fission products to the storage pool. The Draft EIS highlights this concern by pointing out that there has already been an instance of water intrusion into a storage container of EBR-II fuel and subsequent corrosion that lead to a decision to reprocess.

This experience is extended to almost all the other materials proposed for reprocessing. The potential health and safety vulnerability described for unirradiated Mark-42 targets is a cladding breach which results in plutonium oxide being released to the storage basin. The potential for water intrusion to cause corrosion of materials stored in cans is used as the rationale for reprocessing failed or sectioned Tower Shielding Reactor, High Flux Isotope Reactor, Oak Ridge Reactor, and Heavy Water Components Test Reactor fuels, a Mark-14 target, cans of oxide from the Sterling Forest Reactor, and other cans of powdered/oxide fuel which arrive at SRS while the H-Canyon is still operating.

Clearly, if there is indeed an urgent risk, we support efforts to minimize or eliminate that risk. But we also believe DOE has a responsibility in this EIS, as well as in its decision making processes generally, to fully and openly review the facts. Unfortunately, the Draft EIS ignores important information and so provides a misleading argument.

The Final EIS should include a more complete discussion of the options for managing the materials currently proposed for reprocessing. For each material mentioned above, the EIS should address at least the following:

- | | |
|---|-------|
| (1) At the time EBR-II fuel was reprocessed in response to corrosion, why was not all the EBR-II fuel and other material described above reprocessed? | L2-5 |
| (2) What has changed since the decision was made to reprocess the can of corroding EBR-II fuel which makes it urgent to reprocess the fuel and other material before a melt and dilute facility becomes operational? | L2-6 |
| (3) What steps has DOE taken since the discovery of corrosion of some EBR-II fuel to assess and control the hazards associated with the remaining nuclear fuels and materials? | L2-7 |
| (4) What is the evidence that Mark-42 target cladding is likely to breach or that storage cans are likely to leak before a melt and dilute facility becomes operational? | L2-8 |
| (5) If the cladding were to breach or a can were to leak, how much radioactive material would be released into the basin before a melt and dilute facility is expected to become operational? How much of this released material would be captured by the basin's filtration system? How can the filtration system be improved to capture more of the released radioactive material? What are the hazards associated with the releases? | L2-9 |
| (6) What is the estimated risk of continued storage (with and without leakage) until a melt and dilute facility is operational? | L2-10 |

(7) The Draft EIS proposes reprocessing the EBR-II and Sodium Reactor fuels beginning in April or May of this year and the remainder of the material over the next three years. If the EIS is not complete in time to issue a Record of Decision before April, will DOE contend that the situation is so urgent that emergency action to reprocess the EBR-II and Sodium Reactor fuels is warranted? If not, how will delays in completing the EIS affect the proposed schedule for reprocessing these two fuel types and all subsequent materials? How will other possible delays in reprocessing affect plans for the materials discussed in this EIS and other materials proposed for reprocessing?

L2-11

(8) By deferring any or all the reprocessing proposed in this EIS, could money or personnel be made available to expedite planning and construction of a melt and dilute facility?

L2-12

(9) As we understand DOE's current policy, if a container shows signs of leakage and corrosion of the material inside, DOE would move quickly to reprocess the material regardless of the outcome of this EIS. A reasonable alternative which should be added to the EIS is for DOE to continue managing these fuels under this policy until they can be stabilized in the melt and dilute facility.

L2-13

The final item proposed for reprocessing is a core filter block from the Idaho National Engineering and Environmental Laboratory. Here DOE seems to have changed its rationale for reprocessing entirely. For the filter block, DOE contends reprocessing is necessary because it is not cost effective to develop a treatment option for a unique item. The hazard associated with this filter block is not well described in the Draft EIS, and so the reason for taking any action is unclear. Most importantly, though, DOE should refrain from extending the rationale for reprocessing so broadly. Doing so could set a dangerous precedent.

L2-14

L2-15

This precedent may be particularly important as DOE decides how to manage additional nuclear materials not evaluated in the Draft EIS. For example, DOE recently prepared a Processing Needs Assessment but has yet to make decisions regarding how to manage the nuclear materials evaluated in that study. Whether the decision will be based on health and safety criteria, economic considerations, or mere convenience is unknown at this time but could be affected by precedent established in this EIS. The Final EIS should consider this relationship to other decisions, at least by discussing them in terms of cumulative impact and preferably by integrating the decisions.

L2-16

The melt and dilute alternative is proposed for most of the spent fuel and other material discussed in the Draft EIS. In order to safely implement the melt and dilute option, DOE has identified the development of an adequate offgas system as a critical need. The offgas system must be able to capture volatilized radionuclides sufficiently to ensure compliance with relevant standards. We believe DOE's goal, however, should be to use a system capable of limiting releases well below those standards.

L2-17

The history of DOE's efforts to solve technical challenges and to construct and operate new facilities in a timely and cost effective manner gives us pause. In this EIS and throughout the decision making process, DOE should frankly acknowledge past mistakes and the institutional problems that have lead to project failures and to cost and schedule overruns. DOE should also initiate steps to ensure that the development of a melt and dilute facility proceeds along a different

L2-18

path.

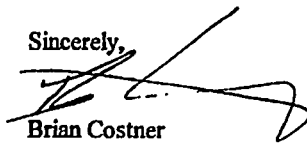
To begin with, DOE should publish in the Final EIS all available information regarding the requirements of an offgas system including the radionuclides of concern, their concentrations, design limitations and other considerations, relevant regulatory requirements, and DOE requirements. Additionally, DOE should open the development and review process to interested people from throughout the DOE system, private industry, academia, and the public. The best chance for success will come from a process which is completely open and which benefits from a dynamic cycle of review, criticism, and improvement.

L2-19

L2-20

Thank you for considering these comments. If you have any questions, please contact me at 2350 10th Avenue East, #113, Seattle, WA 98102 or by phone at 206/329-7394 or by email at bcostner@emeraldnet.net.

Sincerely,



Brian Costner
Consultant

Response to comment L2-1: Comment noted.

EC

Response to comment L2-2: DOE has proposed to use the canyons to stabilize a relatively small volume (about 3 percent by volume and 40 percent by mass) of SNF. Most of the material consists of SNF which is susceptible to releasing fission products in wet storage basins if the storage cans or cladding are breached. The breach of the storage cans and cladding is not a certainty and management of the basin water chemistry is aimed at reducing the probability of releases to the basin. A small amount of other material requires treatment to prepare it for disposal, but poses technical challenges for safe treatment by an alternative non-processing technology. Processing all of these materials through existing facilities in the near term is the most prudent management approach because it would stabilize materials whose forms or types pose a heightened vulnerability to releasing fission products into the basin, or otherwise pose a health and safety concern. See Section 2.4.3.2 of the EIS for this discussion.

Response to comment L2-3: DOE has substantial experience melting nuclear materials and has not identified any reasons why a full-scale operation to melt aluminum-based SNF and dilute the highly enriched uranium would not be achievable. Over the past two years, DOE has conducted research on the melt and dilute process with the specific objectives of reducing technical uncertainties by demonstrating the melt process itself as well as the acceptability for repository disposal of the melted waste form. This effort includes analytical work, actual melting of surrogate materials, and materials tests on the ingots resulting from the melts. These tests have not identified any issues that would lead DOE to believe that the melt and dilute process could not be successfully implemented. DOE also has initiated a pilot program (located in the L-Reactor Area) that will permit melting of irradiated spent nuclear fuel assemblies. The design and construction of a full-scale facility would need to be developed in the context of constrained, out-year budget targets, and funding for such a facility would need to be balanced against other priorities at SRS. DOE

has developed a schedule that can be used as a baseline for near-term planning and budgeting purposes. The FY 2000 funding for SRS has been established, and includes funding for design completion of the pilot scale facility. The pilot scale facility is scheduled to be constructed and placed online by the end of FY 2002.

Recognizing that repository disposal is the ultimate endpoint for the melt and dilute waste form, DOE-SR signed in August 1997 a Memorandum of Understanding with the Nuclear Regulatory Commission (NRC) for their review and feedback on the research effort that DOE is conducting. DOE-SR has provided the NRC several technical reports on the results obtained from the research effort. Based upon their initial review, the NRC in a June 1998 letter stated that "both the direct co-disposal and melt-dilute options would be acceptable concepts for the disposal of aluminum-based research reactor SNF in the repository" (Knapp, 1998). Additionally, as research efforts yield new findings, DOE is providing the information to the NRC for their feedback and review.

Response to comment L2-4: Urgent action to immediately process the vulnerable SNF would be taken in the event DOE identified degraded storage canisters or degraded cladding used to isolate the radioactive material in the SNF from the basin water. DOE has already taken action to improve SNF storage conditions as an interim measure pending SNF disposition. These actions are discussed in Appendix B of the EIS.

The No Action alternative includes the only other action that could be applied to the vulnerable SNF pending the operation of the melt and dilute facility (i.e., repackaging the material in additional storage canisters). However, because the storage and handling operations for this material occur underwater, repackaging would not prevent the contact of potentially reactive material (such as uranium metal) with water in the event of additional container failures.

Response to comment L2-5: DOE felt it was prudent to process the canister of failed EBR-II fuel in the near-term, because of its potential

health and safety vulnerabilities. However, the remaining materials did not present immediate health and safety vulnerabilities so immediate action was not needed. DOE wanted to provide the public an opportunity to comment on disposition of SNF such as the EBR-II as part of the overall planning for SNF management at SRS. Section 1.6 has been revised to discuss this issue.

Response to comment L2-6: Urgent action to immediately process the vulnerable SNF would be taken only in the event DOE identified degraded storage canisters or degraded cladding used to isolate the radioactive material in the SNF from the basin water. DOE proposes to use conventional processing to stabilize some materials before a new treatment facility is in place. The rationale for this processing is to avoid the possibility of urgent future actions, including expensive recovery actions that would entail unnecessary radiation exposure to workers, and in one case, to manage a unique waste form (i.e., core filter block).

Response to comment L2-7: As part of the spent fuel storage program at SRS, the inventory in wet storage is periodically inspected. Also, the water in the storage basins and air in the immediate vicinities of the basins are sampled for heightened radioactivity and radiation levels. These inspections and measurements are used to ensure the integrity of the material being stored.

It was this monitoring protocol that led to the identification in April 1992 of the corroded EBR-II assembly in a storage basin. After this assembly was identified, further inspections of other materials in the storage basin were performed. Those further inspections did not identify any other assemblies requiring immediate processing at that time.

Additionally, DOE has taken the actions described in Appendix B to correct basin vulnerabilities that could cause intact or undamaged SNF to degrade.

Response to comment L2-8: It is not certain that a breach of the Mark-42 cladding or a stor-

age-can leak would occur. However, due to the health and safety concerns that would result from such a release of potentially particulate radioactive material to the SRS storage basins, DOE is proposing to use conventional processing to avoid the possibility of urgent future actions, including expensive recovery actions that also would entail unnecessary radiation exposure to workers. See also response to comment L2-6.

Section 2.4.3.2 has been revised to discuss this issue for the fuel types DOE proposes to process.

Response to comment L2-9: The No-Action Alternative describes how DOE would respond if fuel stored in the basins failed before a melt and dilute facility were operational. The fuel would be processed because the primary engineered barrier to a release of radioactive material would have failed. Based on past operating experience at SRS, if a small number of SNF elements degraded and began to slowly release radioactive material into a basin, the SRS basin filtration systems currently have sufficient capacity to remove the released material to maintain radioactivity concentration in the basin water at an acceptable level. However, the primary purpose of the water in a storage basin is to provide cooling and shielding for the stored SNF and not to confine radioactive material released from failed fuel. DOE considers storing failed fuel a health and safety vulnerability because the primary engineered barrier for radioactive material confinement has failed and the recovery effort would result in increased radiation exposure to workers, and increased waste generation and cost to clean the basin. The filtration and deionization systems are described in Section 2.3.1.

Section 4.1 presents an evaluation of the potential impacts to groundwater from a direct leak to the subsurface from a breach in a storage pool during routine operations. The impacts presented in Section 4.1 of continued storage of SNF in the SRS wet basins include the impacts of processing any material that showed signs of leakage or corrosion.

| EC

TC | Response to comment L2-10: The estimated risk of continued storage until the melt and dilute facility is operational (8 years from now) is 8 times the annual impact in each impact category under the No-Action alternative. Table 4.1-26 presents impacts from the No-Action Alternative for the entire period of analysis (1998-2035). To calculate an annual impact, divide a number by 38 (years in the period for analysis). For example, annual dose to the maximally exposed individual (MEI) from the preferred alternative would be 0.005 millirem (0.19 millirem/38 years). The impacts presented in Chapter 4 for the No Action Alternative include the impacts of processing any material stored in the SRS wet basins that showed signs of leakage or corrosion. Table 4.1-26 has been revised to clarify this.

EC | Response to comment L2-11: The proposed action to process vulnerable SNF would not occur until the Final EIS is completed and the Record of Decision for the EIS has been issued. The Final EIS will be completed in time to issue a Record of Decision that would allow processing of certain fuels through the SRS canyons (if that is DOE's decision) within the time frame of the current canyon operations planning.

Action to immediately process the vulnerable SNF would be taken only in the event DOE identified degraded storage canisters or degraded cladding used to isolate the radioactive material in the SNF from the basin water. This is consistent with prior decisions made following completion of the EIS on Interim Management of Nuclear Materials.

Response to comment L2-12: The proposed processing operations would occur in parallel with other canyon nuclear material stabilization operations. As a result, there would be no personnel available for other work in the event the vulnerable SNF were not processed. Because the canyons already would be operating to process materials not considered in this EIS, there would be no actual cost savings that could be transferred to another activity.

Section 2.6.3 has been revised to clarify this issue.

Response to comment L2-13: DOE evaluated the equivalent of the action proposed by the commenter as the Minimum Impact Alternative. Therefore, the impacts of processing any material that showed signs of leakage or corrosion before DOE fully implemented the Melt and Dilute process have been indirectly analyzed as part of the minimum impact alternative.

Response to comment L2-14: No known health or safety concerns related to the core filter block exist. The core filter block is a unique assembly that includes materials that would not be compatible with the melt and dilute process for aluminum-based SNF. Additionally, the core filter block is composed mainly of depleted uranium and has been exposed to relatively low power. As a result, it contains very little fissile material or fission products. Processing would not extend the time for planned canyon operations, would not generate recovered fissile material, and would produce only a few kilograms of depleted uranium. Therefore, DOE considers processing the core filter block a reasonable action, given the unique character of the item and the very limited amount of material involved. This information is discussed in Section 1.5 of the EIS.

Response to comment L2-15: DOE considers processing the core filter block a reasonable proposed action given the unique characteristics of the item and the very limited amount of material involved. DOE is not extending this rationale for any other materials in this EIS.

Response to comment L2-16: The Process Needs Assessment (PNA; see Section 1.6.2 of the EIS) recommended that the SRS canyons be considered for several materials at various DOE sites. Among those were the Mark-18 and Mark-42 targets at the SRS. These targets have been included in this EIS. For the majority of the other materials, the canyons were identified primarily as a backup alternative.

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DOE is continuing to evaluate the disposition pathways for materials addressed in the PNA to determine which, if any, should be proposed for canyon processing. Prior to making any decisions for the further use of the canyons to support the Department's needs, analysis performed in accordance with the National Environmental Policy Act will be completed. The PNA is a planning document from which proposals for further processing may develop, although no such proposals exist at this time.

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Other material discussed for processing at SRS under the PNA include single-pass reactor SNF at Hanford, a small amount of damaged SNF at Idaho National Engineering and Environmental Laboratory (INEEL), classified fissile material metal parts at the Rocky Flats Environmental Technology Site (RFETS), and plutonium scrap at Hanford. Currently, DOE has no plan or proposal to transfer the single-pass reactor SNF at Hanford or the damaged SNF at INEEL to SRS so that material was not considered for the cumulative impacts under this EIS. In an amended Record of Decision for the *Final Environmental Impact Statement on Storage and Disposition of Surplus Fissile Material*, DOE decided to transfer classified metal from RFETS to SRS for stabilization and storage. DOE is considering transferring the plutonium scrap from Hanford to SRS for stabilization and storage pending appropriate National Environmental Policy Act review. As a result, DOE has included processing that material as part of the cumulative impacts for this EIS.

DOE is continuing to evaluate the inventory of nuclear material at facilities throughout the DOE complex. DOE's Nuclear Material Integration initiative is one such recent effort that has identified material which could be processed at SRS. Although there are no current plans or proposals to process these materials at SRS, DOE considers it appropriate to include a qualitative estimate of impacts as part of the cumulative impacts for this EIS because it is foreseeable that processing at SRS could occur.

DOE has also included processing about 56 MTHM of de-clad sodium-bonded fuel from

INEEL as part of the cumulative impacts analysis in this EIS. Because the *EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (DOE/EIS-0306D) describes processing that material at SRS as a reasonable alternative.

EC

Response to comment L2-17: DOE's objective is to develop an off-gas system that will ensure the safety of workers, the public, and the environment; comply with DOE Orders and Standards; and meet all regulatory requirements. Emissions from any one operating facility are normally a fraction of the applicable requirements. Development of an off-gas system is currently in progress based on results from laboratory experiments and testing. Further testing will be accomplished during the pilot project phase of the melt and dilute system development program that is scheduled to be conducted in fiscal year 2002.

Response to comment L2-18: To reduce technical uncertainties, DOE has conducted several years of research on the process, including actual melting of surrogate materials. A pilot test facility, the L-Area Experimental Facility (LEF), would be constructed in the L-Reactor Area that would permit melting of the irradiated spent nuclear fuel assemblies. The LEF would be used to demonstrate the feasibility of the Melt and Dilute Technology. The design and construction of a full-scale facility would need to be developed in light of the limited available budgets, and funding for such a facility would need to be balanced against other priorities at SRS. DOE has developed a schedule that can be used as a baseline for near-term planning and budgeting purposes. The FY 2000 budget for SRS has been established and includes funding for the completion of the design of the LEF. LEF is scheduled to be constructed and placed online by the end of FY 2002.

EC

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Response to comment L2-19: Regulatory requirements associated with air emissions are discussed in Chapter 3.

Potential emissions from the melt and dilute process are described in Chapter 4 of the EIS.

Radionuclides of concern are described in the document Savannah River Site Spent Nuclear Fuel Environmental Impact Statement Engineering Data Book for Routine Releases, which is listed as Bickford, et. al, 1997, in Chapter 4.

Response to comment L2-20: The development and review process is, in general, open to interested people from throughout the DOE system. For example, development work on the melt and dilute technology is being conducted at Argonne National Laboratory, as well as at the Savannah River Technology Center (SRTC). Briefings on the technology are conducted regularly with the Office of Radioactive Waste Management, and also with DOE and contractor experts from

throughout the DOE complex in the area of management of spent nuclear fuel. Briefings also are conducted for the Defense Nuclear Facilities Safety Board and their staff. The Nuclear Regulatory Commission and its contractors review project documentation and provide DOE-SR feedback on technical issues. In areas where technical uncertainty still remain, such as the design for the off-gas system, DOE will initiate peer review from appropriate technical experts from the DOE complex exclusive of Savannah River Site. For members of the public, DOE will make available, upon request, select summary reports such as the report on off-gas system development to be issued in May 1999. Contact Randy Ponik, DOE-SR (803) 952-2549 or Randall.Ponik@SRS.gov to request additional technical information on technology development activities.

EC

Environmentalists, Inc.
1339 Sinkler Road
Columbia S. C. 29206
(803) 782-3000

Andrew P. Grainger
NEPA Compliance Officer
Savannah River Site
Building 742-A, Room 185
Aiken South Carolina 29802

February 7, 1999

Dear Mr. Grainger:

Researchers with Environmentalists, Inc. (E.I.) have reviewed the Draft Environmental Impact Statement (DEIS) which the Department of Energy (DOE) has prepared regarding Spent Nuclear Fuel Management at the Savannah River Site dated December, 1998. Comments are as follows:

1. Chapter 1. Introduction provides background information, however little is included about problems, such as those associated with plutonium and other difficult-to-handle radioactive materials. This section does discuss uncertainties, such as those that relate to disposal. All twenty-nine (29) references were prepared by DOE, including the Federal Register Notices and EIS's. | L3-1
| L3-2
2. Chapter 2. Proposed Action and Alternatives also has its share of DOE reports as references. A National Academy of Sciences study is discussed, however the information presented is vague and general; not much is included regarding problems or past experiences concerning management of spent nuclear fuel in this country and abroad. | L3-3
3. DOE has failed to include as references the early studies of the SRS by the U.S. Geological Survey and the National Academy of Sciences, yet these documents contain the type of geologic, hydrologic, and meteorological information so needed in Chapter 3. Affected Environment. Instead the agency used primarily its own reports and those of its contractors. Twenty-seven (27) of the sixty-six (66) references were provided by Westinghouse. Other DOE contractors are also represented. | L3-4
(L3-2)
4. It is unclear why the DOE favors a small number of workers being exposed to higher amounts of radiation—external and internal—due to longer time spent in areas where exposures occur rather than using many workers so that individual exposures are lower. This practice is in conflict with the fact that detrimental impacts from exposure to radiation are cumulative. | L3-5

- | | |
|--|---------------------|
| 5. DOE failed to include evidence brought out at licensing hearings before the Nuclear Regulatory Commission (NRC) regarding the Barnwell Nuclear Fuel Plant, a proposed reprocessing facility. These proceedings as well as those related to plutonium and uranium recycling contain sworn testimony which would be useful for DOE to consider in its DEIS, particularly in Chapter 4. <u>Environmental Impacts</u> . Once again, the references identified by DOE are mainly those of its contractors and the agency's own reports. | L3-6

(L3-2) |
| 6. The DOE fails to present the information necessary for understanding the unique problems associated with plutonium and other nuclear materials. For example, the DOE doesn't adequately explain how the operations involving plutonium can be done without the release of particulate matter, including plutonium. What containment provisions have proved to be effective in preventing the release of particulate matter at Nuclear Fuel Services and reprocessing facilities in other countries? How is close to perfect containment maintained during the chopping operation when spent nuclear fuel is being prepared for packaging? | L3-7 |
| 7. Chapter 5. <u>Cumulative Impacts</u> contains numerous assumptions and impacts are minimized. As with previous and later chapters other inadequacies exist such as the limited amount of information regarding actual operating experience. | L3-8 |
| 8. Appendix A. <u>Technology Descriptions</u> discusses various packaging and treatments, such as melt and dilute and chop and dilute. As was true with Chapter 2, the presentations are non-specific and fail to provide information about spent nuclear fuel management in terms of actual operating experience, either in the United States or foreign countries. Only three (3) references are listed for this chapter; two (2) are EIS's prepared by DOE and one is a DOE study. | L3-9

(L3-2) |
| 9. Appendix D. <u>Accident Analysis</u> uses certain assumptions and methods to calculate risks and outcomes from accidents. The computer codes AXAIRO and PRIMUS are identified as being used to make predictions. No actual accidents are discussed in regard to specific information. The references listed relate to calculations and computer models primarily; eleven (11) are identified as reports of Westinghouse, while the remaining ones are regulations and guidance documents. | L3-10

(L3-2) |
| 10. The cumulative effects on the health of persons living in the vicinity of SRS are not given adequate attention in the DEIS. Since it is now known that past radioactive releases from the SRS have been under-reported, there is even more need for the Final Environmental Impact Statement (FEIS) to correct the defects of the draft document. | L3-11 |
| 11. In Appendix G. <u>Public Scoping Comment and Responses</u> , the DOE fails to report fully on what was said by those attending the North Augusta meeting and the letters and e-mails received were not published in the DEIS. It might be possible for E. I. to review the transcript of the meeting and the submittals if this information was available in Columbia rather than only being located in the Aiken area. | L3-12 |
| 12. The two reference documents to the DEIS, regarding costs and non-proliferation, include some footnotes calling attention to the groups that developed cost projections, identified preferred alternatives and other similar matters based primarily on calculations rather than on historic records. Although one report states that "an | L3-13 |

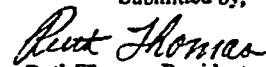
independent study" was commissioned, apparently much of the work was done by the Westinghouse Savannah River Company.

(L3-2)

13. E. I. has members who live close to Interstate Nuclear Services (INS), the company that cleans the contaminated clothing and devices from the SRS. At the public hearing in Columbia on January 28th, questions were raised about INS's nuclear laundry as well as on other subjects related to the DEIS. These questions and the comments made need to be presented fully in the FEIS along with the publication of all written submittals, including this letter. E. I. appreciates DOE's holding a hearing in Columbia last month and members look forward to future occasions for E. I. and other organizations and persons in the Columbia area to meet and talk with DOE representatives.

L3-14

Submitted by,



Ruth Thomas, President
Environmentalists, Inc.
1339 Sinkler Road
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Response to comment L3-1: The SRS manages a variety of nuclear materials including plutonium, uranium, and SNF. The purpose of this EIS is to discuss the management of SNF at the SRS. The introduction describes why it is necessary to take care of this spent nuclear fuel. Unlike commercial SNF, these fuels are made of highly enriched uranium (HEU) and therefore present proliferation considerations. Also, they may not be acceptable for direct disposal in a geologic repository as can commercial fuels because the HEU makes criticality a possibility if the storage configuration were changed. Another difference between SNF and commercial fuel is the physical form and the location of the fission products. A large portion of the SNF is metal or metallic alloy which could corrode and release the intermixed fission products. Commercial SNF is formed as a ceramic oxide and is naturally more resistant to corrosion. Section 1.5 of the EIS describes the material evaluated in this EIS and includes a discussion of the potential health and safety vulnerabilities that may exist for a relatively small volume (about 3 percent by volume and 40 percent by mass) of the SNF. Additionally, Appendix B of the EIS describes vulnerabilities that DOE has identified with SNF management facilities at the SRS, the corrective actions for the vulnerabilities, and the status of the actions.

Plutonium and other radioactive materials are outside the scope of this EIS. DOE has prepared other NEPA evaluations describing and evaluating issues and concerns related to storing, processing and handling plutonium and other materials at SRS. These documents include the F-Canyon Plutonium Solutions EIS (DOE/EIS-0219, 1994) and the Interim Management of Nuclear Materials EIS (DOE/EIS-0220, 1995). NEPA documents that discuss ongoing or proposed actions at SRS, including actions that involve plutonium, are described in Section 1.6 of this EIS.

Response to comment L3-2: DOE seeks to use the best available technical information when preparing EISs. DOE is currently the only U.S. government agency that is managing aluminum-based SNF; therefore, much of the technical in-

formation on this program would necessarily have been prepared by DOE or its contractors. In addition, DOE used actual historic SRS data when preparing this EIS (e.g., for SNF wet basin impacts and impacts of canyon operation). DOE requested an independent evaluation by the National Academy of Sciences of DOE's aluminum-based SNF disposition technical program. This independent review was completed in 1998 and the results and recommendations are summarized in Section 2.6 of the EIS.

Response to comment L3-3: Section 2.6 of the Draft EIS discussed the recommendations of the National Academy of Sciences in regard to the selection of a SNF treatment technology. The discussion has been updated to describe DOE actions in regard to the recommendations.

Appendix B of the EIS describes vulnerabilities that DOE has identified with SNF management facilities at the SRS, the corrective actions for the vulnerabilities, and the status of the actions. DOE evaluated the management of foreign research reactor SNF in the U.S. under the Final EIS of a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (DOE/EIS-0218F, February 1996).

Chapter 2 of this EIS describes the actions DOE proposes to take to manage SNF at SRS and the existing and proposed facilities required to do so. A National Academy of Science Study on treatment options for spent nuclear fuel is referenced, as is Defense Nuclear Facility Safety Board Recommendation 94-1 regarding health and safety vulnerabilities of spent nuclear fuel storage and DOE's own study of the same issues.

Section 2.7 has been updated to describe DOE's actions in regard to the recommendations.

Response to comment L3-4: DOE makes an effort to describe the affected environment on the basis of the most current information available. More recent studies that were used to prepare the EIS incorporate all relevant information from these earlier documents.

Response to comment L3-5: DOE and the SRS operating contractor, Westinghouse Savannah River Company (WSRC), limit worker exposure to 500 millirem per year, a dose below which health impacts are not expected to occur. See Section 3.7.3 Worker Radiological Health for more information. In addition, on a task-by-task basis, extensive effort is taken to limit exposure to As Low As Reasonably Achievable (ALARA). The ALARA program balances benefits – performing the task – and risks – exposure to radiation. It is not DOE or WSRC policy to use smaller numbers versus larger numbers of workers, as the commenter suggests.

TC Please note that spreading the potential radiation dose among a larger group of workers, as the commentor suggests, would not reduce the estimated risk of a radiation-induced cancer fatality among exposed workers. Using more workers might reduce doses to individual workers, but more workers would be exposed – the collective dose and overall estimated risk for workers would remain the same. This result is based on the assumption that there is no threshold level below which exposure to radiation poses zero risk. See section 4.1.1.3.1 (page 4-11) for further information about radiological health effects.

Response to comment L3-6: The proceedings mentioned by the commenter were from the early 1970s. DOE prefers to use the most recent scientifically validated data available. In addition, Barnwell Nuclear Fuel Plant never operated and did not process nuclear material. DOE prefers to use data from operating facilities. The extensive database of SRS operating experience has been used in this EIS to assess the expected impacts of the operation of existing facilities, and this same database has been applied to estimate the potential impacts of operation of proposed spent nuclear fuel management facilities.

Response to comment L3-7: Section 2.3 of the EIS has been revised to include a description of radioactive material containment techniques that are or would be applied.

Response to comment L3-8: It is difficult to respond to this comment because the commenter does not identify the assumptions or impacts questioned. However, it was not and is not DOE's intent to minimize the potential impacts of ongoing or proposed operation of facilities or programs described in the cumulative impacts section (Chapter 5). For existing facilities, impacts are based on actual operating experience (e.g., air emissions data taken from the SRS Annual Environmental Report). For proposed facilities described in the cumulative impacts section, data have been taken from the EISs prepared for those facilities. In cases where a program is proposed, like plutonium disposition, impact information is based on currently available design information because the facilities do not exist.

Response to comment L3-9: The extensive database of SRS operating experience has been used in this EIS to assess the expected impacts of the operation of existing facilities, and this same database has been applied to estimate the potential impacts of operation of proposed spent nuclear fuel management facilities.

Response to comment L3-10: As indicated in Appendix D (page D-1, second column) DOE based the frequencies of potential accident scenarios at existing facilities on safety analyses and historical data about event occurrences. For proposed new facilities without design details, DOE based the accident frequencies on hazard analyses, historical data for similar facilities and operations, and best estimates. Most of the postulated accidents have a very low probability of occurring and have never happened at SRS. Nevertheless, conservative assumptions (i.e., tending to overstate the risk) were utilized to ensure that the EIS analyses impacts of the potentially most severe reasonably foreseeable accidents. DOE has provided estimates of their probability and of the impacts should they occur. In the analyses, no credit is taken for the safety systems DOE includes in the facilities to ensure that these severe accidents never occur.

Response to comment L3-11: The EIS describes the projected impacts of DOE's SNF manage-

ment alternatives on the health of site workers and the public. In addition, Table 5-4 describes the projected health effects from current and reasonably foreseeable SRS operations on the offsite population surrounding SRS and on site workers. These projected impacts are based on actual release data from current or recent past operations.

The recent draft report by the Centers for Disease Control and Prevention (CDC) concludes that emissions of certain radionuclides from the Savannah River Site were underreported, primarily in the 1950s and early 1960s. The CDC report states that its estimates for the time since the late 1960s are in close agreement with DOE estimates made at the time. Neither the CDC nor DOE has data on the cumulative health impacts. Phase I of the study developed releases numbers. Phase II of the study will look at dose affects. Copies of the report can be obtained from the CDC (Paul Renard, 4770 Buford Highway NE, M.S. F-35, Atlanta, GA, 30341 or by phone at 770-488-7030).

Response to comment L3-12: DOE believes that the description of public scoping comments received by letter, email, and at the scoping meeting is an accurate summary that captures the sense of the comments provided to DOE. No comments or issues were omitted from Appendix G of the Draft EIS. Copies of the transcripts can be requested from DOE at 1-800-881-7292.

Response to comment L3-13: The nonproliferation study was prepared by DOE's Office of Nonproliferation and National Security in Washington, D.C., and, by necessity is a qualitative evaluation based on the experience of the authors regarding the application of U.S. nonproliferation policy. The study was reviewed by

a panel of experts independent of DOE: Matthew Bunn, Belfor Center for Science and International Affairs, Harvard University; Frank von Hippel, Woodrow Wilson School of Public and International Affairs, Princeton University; George Bunn, Center for International Security and Cooperation, Institute for International Studies,

Stanford University; Harold Bengelsdorf, Bengelsdorf, McGoldrick and Associates, LLC; and David Albright, Institute for Science and International Security.

The cost study was prepared by the DOE Savannah River Operations Office based on actual facility operating data where possible. In instances where data on actual facility operations experience could not be obtained, estimates were made based on experience with similar facilities or on preliminary or conceptual design.

It was reviewed by Lawrence Buc, Project Performance Corporation; Kenneth Bulmahn, Sandia National Laboratories; Sander Glick, Project Performance Corporation; William Hurt, INEEL; and John McConaghy, Duke Engineering and Services, Inc.

Response to comment L3-14: In regard to the INS laundry facility, see DOE's response to comment L12-2.

DOE has published comment letters and emails containing comments on the draft EIS in this Final EIS. DOE has chosen to summarize comments received during the public meetings rather than print the entire transcripts. The transcripts are available for inspection at the Gregg-Graniteville Library, University of South Carolina - Aiken and at the Thomas Cooper Library, University of South Carolina, in Columbia.



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February 11, 1999

Mr. Andrew Grainger
NEPA Compliance Officer
U.S. Department of Energy
Savannah River Operations Office
Building 742A, Room 183
Aiken, SC 29802

**Comments on the Department of Energy's
Draft Environmental Impact Statement on
Spent Nuclear Fuel Management**

Dear Mr. Grainger:

The Nuclear Control Institute (NCI) welcomes the opportunity to comment on the draft EIS issued by the U.S. Department of Energy on how to manage spent nuclear fuel, targets, and other material now stored at the Savannah River Site (SRS). NCI is supportive of those aspects of the Draft EIS which seek to implement a policy to avoid reprocessing of nuclear materials at SRS by development of alternative stabilization methods that do not result in separation of weapons-usable material. We congratulate the Department for issuance of a document with this stated goal.

L4-1

NCI believes that DOE has made progress in implementing non-reprocessing alternatives, as decided in the 1996 Record of Decision (ROD) on the *Final EIS on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor SNF*, and is pleased to see that DOE has devised a non-reprocessing plan to deal with HEU-bearing aluminum-clad spent fuel. We urge the Department to stand by its current assessment that "DOE has greater confidence now than at the time the Foreign Research Reactor EIS's Record of Decision in the feasibility and availability of non-chemical reprocessing technologies" (Draft EIS, p. S-6). We also urge further progress toward implementation of such non-chemical processing technologies once the ROD for the *SRS Spent Nuclear Fuel Management Final EIS* is issued.

L4-2

Selection of the "Melt and Dilute" technique for 97% (in volume) of this material presents a solution which is stable for geologic disposal and unattractive for nuclear weapons use, as it will result in ingots which do not contain highly-enriched uranium (HEU). The alternative of reprocessing the spent fuel in the H-Canyon at SRS would produce purified, weapon-usable HEU as an intermediate product and extend the life of the aging plant for anywhere from five to thirty years, posing a continued and unacceptable proliferation and environmental risk. Development of Melt and Dilute will

L4-2a

Strategies for stopping the spread and reversing the growth of nuclear arms.

Paul L. Leventhal, President, Peter A. Bradford, David Cohen, Julian Koenig, Sharon Tanzer, Roger Richter, Dr. Theodore B. Taylor
BOARD OF DIRECTORS

also provide a technological alternative which can be used by other countries to manage their own aluminum-clad HEU spent fuel.

Although the largest volume of material -- representing 60% in mass of materials considered in the EIS -- is proposed to be dealt with using the Melt and Dilute method, we are discouraged that DOE is considering reprocessing of 40% of materials by mass -- primarily declad EBR-II uranium metal spent fuel. DOE's claim that such treatment is necessary because it "presents a potential health and safety vulnerability, is in a form that may be unacceptable for placement in a geologic repository, and/or for which the development of a second non-reprocessing technology would be too expensive" (Draft EIS, p. S-15) should be re-examined. Given that maintaining of the canyons in an operable mode is exorbitantly expensive, we are convinced that a concerted DOE effort to develop non-reprocessing alternatives, perhaps as simple as placing EBR-II canisters in an additional corrosion-resistant sleeve, could yield results both technologically and economically efficient.

L4-3

While DOE states in the Proposed Action that operation of the canyons for reprocessing of spent nuclear fuel would not be extended beyond the closure target date of "about 2005," little assurance is given in the EIS that this date will be met. We are concerned that DOE will once again allow the date for closure of the canyons to slip, thereby providing an opening for new reprocessing missions to be developed. Along with issuance of the ROD, DOE should make a clear commitment that the canyons will be closed within the "current planning basis" and that no slippage of that deadline will be tolerated. Given the commitment to the Melt and Dilute technique, it is now time for DOE to present a firm closure date for the SRS reprocessing canyons.

L4-4

DOE's contention in the EIS that operation of the canyons is needed for stabilizing nuclear materials that present health and safety vulnerabilities carries an undesirable message with potentially negative ramifications for international non-proliferation interests. Other nations engaged in or planning for reprocessing can use this same justification to continue operation of their own facilities, with resultant separation of weapons-usable material. Thus, we continue to question the soundness of DOE's reasoning in making the stabilization argument. While health and safety considerations must be taken into account in handling the materials at SRS, DOE should not allow the canyons to become a technology in search of a use.

L4-5

Avoidance of use of reprocessing facilities at SRS will result in a reduction of high-level waste solution pumped into storage tanks at the facility (for which, due to the failure of the In-Tank Precipitation method, there is no viable stabilization path at the present time). It will also send a signal internationally that viable alternatives exist to reprocessing of such material. Also, early shutdown of the SRS canyons will be seen internationally as a good-faith disarmament effort and will facilitate negotiations and eventual implementation and verification of a ban on production of fissile materials.

L4-6

We support the goal of having the Melt and Dilute process in operation by the year 2005, but DOE should establish a concrete timeline of technological milestones and funding which would make this happen. As the technology is within reach, DOE should insure that full funding for the Melt and Dilute technique is in the budget for FY 2000 and in subsequent years. Use of existing facilities associated with the L-Reactor is seen by us as a practical and efficient way to proceed as long as that those facilities meet the most stringent health and safety standards. In this regard, we are encouraged by DOE's assessment that an off-gas system to capture fission products volatilized during melt-and-dilute treatment "could be designed using proven approaches for managing off-gases" (Draft EIS, p. S-18), and we urge DOE to provide adequate funding to support a focused effort to design and demonstrate such as system.

L4-7

L4-8

We also support the plan presented in the draft EIS to transfer to dry storage 20 MTHM of non-aluminum clad spent fuel now stored in the Receiving Basin for Offsite Fuel (RBOF). We applaud the associated goal of phasing out the use of RBOF by 2006, and thus would like to see active development of the L-Reactor Disassembly Basin to which aluminum-clad fuel could be transferred. The ROD should also include a commitment to a closure of RBOF within a specified period, thus underscoring the commitment to development of alternative storage and handling techniques for materials now stored in that basin. Establishing a timeline for development of alternative storage and management techniques will help to ensure the RBOF is emptied and decommissioned by a date certain, along with the associated reprocessing facility.

L4-9

L4-10

L4-11

It is clear that the end of the operational lives of the SRS reprocessing canyons is now in sight. The stated strategy and preference of DOE in the draft EIS "to use non-chemical separation processes when practical" (Draft EIS, p. S-6) should facilitate their shutdown in a time frame consistent with current planning basis (by about 2005), provided adequate resources are made available to fully develop the technology alternatives. The final EIS should make a clear commitment to ensuring the availability of the necessary funding resources.

L4-12

We summarize our comments on and requests for revisions in the draft EIS as follows:

1) DOE's decision to implement the Melt-and-Dilute process for HEU spent fuel is technologically sound. Its implementation should be fully funded.

L4-13

2) DOE should present its timeline for implementation of Melt-and-Dilute and keep the public informed as progress toward that goal is made and give explanation if delays are encountered. Development of the system, including off-gas treatment, should be carried out under public review and through application of highest technological standards.

L4-14

3) DOE needs to present a better analysis of costs of reprocessing and of reprocessing alternatives in the final EIS. This analysis must include costs of current

L4-15

operation on the canyons, including cost of waste management and cost per unit of material reprocessed.

4) For any reprocessing which DOE includes in the final EIS, DOE should more clearly define actual health and safety risks associated with materials designated for reprocessing and should commit not to reprocess if non-reprocessing technologies could be effectively employed.

L4-16

5) In the final EIS, DOE should describe other pending actions and decisions which might impact any decision to reprocess spent fuel at SRS. For example, if DOE were to import materials not described in the RERTR EIS or to consider reprocessing of spent fuel or fissile materials in storage at DOE sites, then DOE should candidly describe any use of the canyons it might envisage in handling such materials. An assessment of how this would affect the "current planning basis" underlying the spent fuel EIS should be made and justification should be given for use of the canyons versus non-reprocessing alternatives.

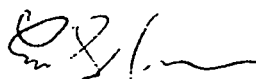
L4-17

L4-18

Sincerely,



Tom Clements
Executive Director



Dr. Edwin Lyman
Scientific Director

attachment: 2-page NCI news release, December 28, 1998

Response to comments L4-1, L4-2, and L4-2a:
Comments noted.

Response to comment L4-3: The volume of SNF DOE is proposing to process is relatively small (i.e., about 3 percent by volume and 40 percent by mass of the SNF evaluated under this EIS). This small volume of SNF has a large mass because it is almost all uranium metal. Most of the SNF in this EIS consists of aluminum alloyed with uranium which has a high volume to mass ratio compared to uranium metal.

The canyons at SRS are currently operating to stabilize materials outside the scope of this EIS. The operations required to stabilize the SNF proposed for processing are not expected to extend canyon operating time beyond the existing planning basis. Because the canyons are operating, their utilization offers the most cost-effective solution to stabilizing these materials. DOE believes that processing the proposed material through existing facilities is appropriate based on an approach of stabilizing materials before they become an actual health and safety concern.

The only interim measure available would be repackaging. The EBR-II SNF currently is overpacked in corrosion-resistant storage canisters. One of these canisters already has leaked. DOE could repackage the EBR-II SNF in the storage basins (this option is included as part of the No Action Alternative in the EIS) but, since repackaging operations would occur underwater, repackaging would not prevent potentially reactive material (such as uranium metal) coming in contact with water in the event of additional container failures.

Response to comment L4-4: The current canyon planning basis schedule includes processing the SNF proposed under this EIS. Decisions regarding action to close the canyons and decontaminate and decommission them are beyond the scope of this EIS. However, DOE expects to shut down the canyons after nuclear material

stabilization activities that involve the canyons have been completed.

Response to comment L4-5: DOE considers processing the SNF for which a potential health and safety vulnerability could exist a prudent action to avoid the release of radioactive materials into the waters of the SRS basins. DOE evaluated the nonproliferation implications of canyon processing operations in a report that is discussed and referenced in Section 2.6 of the EIS.

Processing of the material in the canyons is not being performed for the recovery of weapons-grade plutonium. The EIS is clear that the plutonium that comes out of the Group A fuel stabilization process will be dispositioned in accordance with decisions reached after completion of the Surplus Plutonium Disposition EIS. Highly-enriched uranium will be blended to low-enriched uranium in the canyon processing.

Response to comment L4-6: DOE's strategy is to select a new non-chemical processing technology that would put SNF, including foreign research reactor SNF, into a form or container suitable for direct placement in a geologic repository. DOE's preferred alternative is to treat the majority of SNF using a non-separations-based technology, i.e., melt and dilute. DOE proposes to apply the Conventional Processing technology only to a relatively small volume of SNF at SRS (about 3 percent by volume and 40 percent by mass). Not processing this material would not result in a significant change to the canyon operating schedule. The volume of high-level waste produced by processing this small volume of SNF is very small compared to the volume of high-level waste currently being managed at SRS. The process for treating the high-level waste salt fraction is the subject of another EIS.

| EC

| EC

Response to comment L4-7: A timeline has been established for the development, design and implementation of the melt and dilute facility. This timeline will be controlled through

DOE's line item budget and procurement process pending completion of this NEPA process. The design and construction of a full-scale facility would need to be developed in the context of constrained, out-year budget targets, and funding for such a facility would need to be balanced against other priorities at SRS. DOE has developed a schedule that can be used as a baseline for near-term planning and budgeting purposes. The FY 2000 budget has been established and includes funding for design completion of the L-Area Experimental Facility (LEF). The LEF is a pilot test facility which will demonstrate the feasibility of the melt and dilute technology. LEF is scheduled to be constructed and placed online by the end of FY 2002.

Response to comment L4-8: As stated in DOE's response to comment L2-17, the Department's objective is to develop an off-gas system that will ensure the safety of workers, the public, and the environment; comply with DOE Orders and Standards; and meet all regulatory requirements. Emissions from any one operating facility are normally a fraction of the permissible emissions. Development of an off-gas system is currently in progress based on results from laboratory experiments and testing. Further testing will be accomplished during the pilot project phase of the melt and dilute system development program.

Response to comments L4-9 and L4-10: Comments noted.

Response to comment L4-11: The expected timelines for actions related to facility start-up, operating schedules, or potential shutdown are discussed in the descriptions of the SNF management technologies evaluated in the EIS. The preferred alternative in the Final EIS contains a discussion of these activities as applicable. The Record of Decision will contain a description of the selected alternative including a description of planning-basis dates when certain activities, such as the phase out of the Receiving Basin for Offsite Fuel, are expected to occur.

Response to comment L4-12: See response to comment L4-7.

Response to comment L4-13: See response to comment L4-7.

Response to comment L4-14: See responses to L4-7, L4-8, L2-18, and L2-20.

Response to comment L4-15: Section 2.6 of the EIS presents the results of an evaluation of all SNF management technologies considered in the EIS.

The portion of canyon operating costs associated with processing the potentially vulnerable SNF is small. DOE estimated these costs to be less than 3 percent of the cost that DOE will spend to operate the canyons for the stabilization of other nuclear materials. Additionally, the SNF processing cost is, in reality, a construct for estimation purposes based on allocating a cost to a dissolver run. DOE would not expect to save that cost if the SNF analyzed in this EIS were not processed nor would DOE expect that the funding would be available for use in other programs. The processing cost estimates were based on current canyon operating costs including the cost of waste management. Section 2.6 has been revised to clarify this information.

Response to comment L4-16: The primary potential health and safety risks associated with storage of SNF in wet basins relate to the possibility of dispersion of radioactive material from the SNF to the basin water. If this were to occur, it could impact criticality controls in the basins, result in increased radiation exposure of operations personnel, and result in the generation of increased quantities of radioactive waste that must be managed. In addition, fuel retrievability and handling would be adversely affected. The health and safety risks of continued wet storage of SNF are addressed in Sections 4.1.1.3, 4.1.2, and 4.2 of the EIS.

Response to comment L4-17: See response to comment L2-16.

Response to comment L4-18: See response to comment L2-16.



Fred E. Humes
Director

February 8, 1999

Andrew R. Grainger
NEPA Compliance Officer
U.S. Department of Energy
Savannah River Operations Office
Building 742A, Room 183
Aiken, South Carolina 29802

Dear Mr. Grainger:

On January 28, 1999 my organization provided comments on the Spent Nuclear Fuel Management Draft Environmental Impact Statement. We support the DOE objectives of (1) safe management for spent nuclear fuels (SNF) at the Savannah River Site (SRS) and (2) preparing the fuels for final disposition in the National Repository. We are especially anxious to assure that the fuels are prepared for shipment to the National Repository to reduce the potential for a long-term safety and health legacy at the SRS.

L5-1

L5-2

In reviewing your Draft EIS, it is not apparent if the DOE has adopted a treatment technique that will be acceptable to the National Repository. It is our position that the proposed treatment should provide an equal or greater assurance of acceptance in the National Repository when compared to Commercial Spent Nuclear Fuel or DWPF vitrified waste. Absent such assurance, the option of managing SNF by chemical processing and vitrification must be maintained. The worst case scenario for the citizens of South Carolina and Georgia is that (1) the alternate treatment technology is not acceptable to the National Repository and (2) that the SRS chemical separation capability has been abandoned. The DOE programmatic strategy should mitigate against this circumstance and the EIS must address its impacts.

L5-3

In preparing the Final EIS, please address the following three items:

- the program for assuring that the alternate treatment technologies will provide a product acceptable for receipt in the geologic National Repository, and its acceptability compared to Commercial Spent Nuclear Fuel and DWPF vitrified waste,

L5-4

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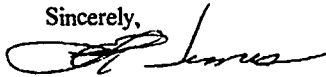
- a commitment from the DOE that SRS SNF will be treated by chemical separations and DWPF if the alternate technology program does not provide a waste form acceptable to the National Repository, and
- a description of the safety and environmental impacts if the alternate treatment program is not successful, the chemical separations capability is not available and the SNF must be managed for the long term without an approved disposition strategy.

L5-5

L5-6

A copy of our earlier comments is attached for your information. Thank you very much for the opportunity to comment on this Draft EIS.

Sincerely,



Fred E. Humes
Director



Fred E. Humes
Director

Statement for the Record
Savannah River Site Spent Nuclear Fuel Management
Draft Environmental Impact Statement
January 28, 1999

My name is Ernest Chaput and I represent the Economic Development Partnership of Aiken and Edgefield Counties, South Carolina. For the past three years my organization has evaluated the impact of the Savannah River Site and its missions upon the economic base and community expectations within the Central Savannah River Area. As part of our activities we have commented upon proposed DOE actions as they relate to SRS capabilities, safety and environmental considerations and economics.

The objective of Spent Nuclear Fuel Management Draft EIS is very significant. Spent Nuclear Fuel (SNF) currently on the SRS and projected to be received over the next 35 years is the only significant post-cold-war safety and environmental legacy which does not have an approved path for long-term stabilization and ultimate disposition. Unless properly managed and disposed, these fuels have the potential to affect the safety of SRS workers and the general public and to adversely impact the environment. It is incumbent upon DOE to address ultimate disposition so that we do not leave a legacy of waste to be addressed by a future generation. DOE has an especially paramount obligation to assure the safe disposition of foreign fuels that are being shipped to the Savannah River Site over the objections of the citizens of South Carolina.

DOE appears to recognize these needs as it describes the decisions and proposed actions to be based on this EIS; including "safely manage SNF" and "packaging aluminum-based fuel for offsite shipment and placement in a monitored geologic repository." DOE has previously determined that non-aluminum-based SNF would be transferred to its Idaho site. When implemented these actions would achieve our objective of assuring that a disposition path exists for all SRS spent nuclear fuel, and that we do not leave a legacy of interim storage to our children. Our comments are provided in the context of this background and expectation.

We have two major comments on the Draft EIS:

Comment Number 1 - Waste Form Acceptance into the National Repository

The Draft EIS does not describe an assured path for disposition of aluminum-based spent nuclear fuels to the National Repository. Two deficiencies are present:

| L5-7

- a. Waste Acceptance Criteria for placement of SNF into the national repository are not available. The EIS states that a systems requirements document has been prepared, and that DOE is "working" with the Nuclear Regulatory Commission and the Yucca Mountain office to establish treatment technology specifications and identify waste form criteria and waste characterization requirements.

L5-8

- b. The Draft EIS does not assess the performance of each alternative management technology option against the current waste acceptance system requirements or potential waste acceptance criteria. Rather, the Draft EIS makes an assumption that each technology can prepare a "road ready" package which is acceptable to the geologic repository. The Draft EIS does not include information on the relative conformity of individual technology waste forms with potential waste performance requirements.

L5-9

This is a serious deficiency. The ultimate objective of the EIS is to prepare spent nuclear fuel for placement in the national repository, yet the ability to evaluate the alternative technologies is not available.

At the present time only the conventional processing alternate has the capability for final disposition of spent nuclear fuel - yielding surplus uranium and plutonium and vitrified high level waste. DOE has established disposition strategies for surplus uranium and plutonium; and waste acceptance criteria have been developed for the vitrified waste form with the existing DWPF process meeting those requirements.

The Secretary of Energy has directed that reprocessing operations to produce strategic nuclear materials be phased out, with conventional reprocessing being used only to stabilize fuels with potential health and safety vulnerabilities. The Draft EIS states that reprocessing facilities will not be available after the year 2005.

The Draft EIS does not state when final waste acceptance criteria for SNF will be available, nor does it state when the definitive acceptability of the waste form from the alternative technology options will be ascertained. **There is no discussion or assurance that the any of the alternate SNF management technology options will meet national repository acceptance criteria before existing SRS conventional reprocessing capabilities are shutdown. If none of the alternate technology options can meet repository acceptance criteria and if the existing reprocessing canyons have been shut down, a significant health, safety and environmental risk will eventually exist at the SRS.**

L5-10

L5-11

L5-12

This deficiency must be corrected. We offer three possible suggestions:

1. That the EIS be modified to include (1) final SNF acceptance criteria for placement in the national repository and (2) an evaluation of each management technology option against the final acceptance criteria. Only those options with a

L5-13

level of acceptance certainty similar to DWPF vitrified waste should be selected for management of SNF at the Savannah River Site, and

2. That DOE maintain the operability of the SRS reprocessing canyons until a technology option(s) has been identified and demonstrated for management of SRS SNF which yields a product that meets national repository acceptance criteria.
3. If DOE cannot provide assurances that the accepted reprocessing/DWPF capability will remain until a proven alternate is selected, then the EIS needs to be modified to include the health, safety and environmental impacts of long term interim storage of SNF at SRS, including the periodic repackaging to maintain integrity and a long-term resolution for safe storage or disposition of long-life fission products and actinides.

L5-14

L5-15

Comment Number 2 - Inclusion of Uranium Credit Cost Data

The Draft EIS and its supporting Alternatives Cost Study should be modified to include the value of uranium credits for each option. There is very little difference in cost between the less complicated technology options and the effect of potential credits would provide useful information. It is acknowledged that the uranium market is currently in flux, but credits will occur over a twenty year period when markets may be more settled.

L5-16

Thank you very much for the opportunity to comment on your draft EIS and I look forward to your resolution of these comments.

Response to comments L5-1 and L5-2: Comments noted.

Response to comment L5-3, L5-5, L5-8, L5-9, L5-10, and L5-11: DOE-SR is working closely with the U.S. Nuclear Regulatory Commission (NRC - the Federal agency that would license the operation of a geologic repository) to ensure that the final product from the selected SNF treatment technology would be acceptable for disposition. Under current planning, DOE would submit a license application to the NRC in 2002 (if the Yucca Mountain site is found suitable and DOE receives other authorizations required under the Nuclear Waste Policy Act) to support the repository safety demonstration. The acceptance criteria at that time would be the basis for the waste acceptance specifications in the license application. This is before the melt and dilute facility would become operational and before the canyons cease operating. Final waste specifications will not be available until after NRC approves construction of a repository and authorizes a license for DOE to receive and store SNF and high-level waste, prior to repository operations. If the site is suitable and licensed, DOE plans to begin repository operations in 2010. The EIS has been revised to discuss in greater detail expected repository acceptance criteria and compare the treatment technology products to that criteria. This information is discussed in Section 2.2.1.

Recognizing that repository disposal is the ultimate endpoint for the melt and dilute waste form, DOE-SR signed in August 1997 a Memorandum of Understanding with the Nuclear Regulatory Commission (NRC) for their review and feedback on the research effort that DOE is conducting. DOE-SR has provided the NRC several technical reports on the results obtained from the research effort. Based upon their initial review, the NRC in a June 1998 letter (Knapp 1998) stated that "both the direct co-disposal and melt-dilute options would be acceptable concepts for the disposal of aluminum-based research reactor SNF in the repository." Additionally, as research efforts yield new findings, DOE is providing the information to the NRC for their feedback and review.

The current canyon operating schedule extends until fiscal year 2006. DOE expects to begin operating the new SNF treatment facility within the decade.

Response to L5-4: No waste form has been declared acceptable for a geologic repository. Until an operating license is granted, uncertainty exists for all waste forms.

Response to comment L5-6: The scenario described in the comment (long-term SNF management without an approved disposition strategy) is equivalent to the No-Action Alternative analyzed in the EIS. The No-Action Alternative assumes that the SNF would remain in storage in SRS wet basins over the entire 38-year period of analysis.

Response to comment L5-7: DOE is confident that the preferred alternative described in the EIS does provide an assured path for disposition of aluminum-based spent nuclear fuels in the geologic repository. Preliminary repository waste acceptance criteria have been established by DOE's Office of Civilian Radioactive Waste Management. While preliminary, the repository requirements are well understood by DOE and will be finalized during NRC's review of the repository license application. These criteria form the basis for the technology comparisons made in this EIS. The final EIS (Section 2.2.1) compares the waste forms that would be produced by Melt and Dilute and Direct Disposal to the components of the repository waste acceptance criteria. DOE proposes to use the melt and dilute technology to stabilize the largest volume of material considered by this EIS. Technical bases have been developed to establish functional performance requirements for a melt and dilute treatment and storage facility.

Response to comment L5-12: Health, safety, and environmental risks associated with continued storage were estimated for the No-Action Alternative.

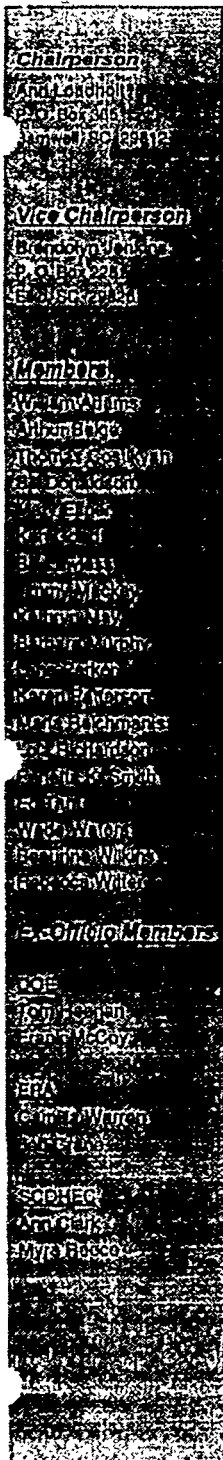
Response to comment L5-13: The comments are beyond the scope of the EIS.

Response to comment L5-14: Under the current planning basis DOE will have high confidence in the acceptability of the waste form before the canyons cease operations.

Response to comment L5-15: The situation described in the comment was evaluated under the No-Action alternative.

Response to comment L5-16: The cost study referenced in the EIS states that uranium credits

in the range of \$110 million to \$150 million were considered in developing the cost report. However, due to the recently signed agreement between Russia and the United States that calls for the potential deferment of HEU sales, no actual value was assigned in the report for calculating life-cycle costs associated with processing activities. Section 2.6.5 of the EIS has been revised to clarify this point.



Savannah River Site

CITIZENS ADVISORY BOARD

A U.S. Department of Energy Site-Specific Advisory Board

January 27, 1999

Mr. Greg Rudy, Manager
U.S. Department of Energy
Savannah River Operations Office
P.O. Box A
Aiken, S.C. 29808

Dear Mr. Rudy:

On behalf of the SRS CAB, I am pleased to forward you four of five recommendations adopted at our January 25-26, 1999, meeting held in Hilton Head Island, S.C. Recommendation 74 addresses the SRS Spent Nuclear Fuel Draft EIS. Recommendation 75 regarding Interim Corrective Measures at the Southwest Plume from the Old Radioactive Waste Burial Ground will be transmitted under separate cover and accompanied by a minority report. Recommendation 76 addresses Plug-In Records of Decision at SRS. We are requesting that you forward Recommendation 77 regarding the Waste Isolation Pilot Plant RCRA Part B permit to DOE-Carlsbad and the New Mexico Environmental Department by January 29, 1999, before the public comment period ends. Recommendation 78 addresses plans to conduct an EIS on High Level Waste Tank Closures.

We would appreciate your written response regarding all of the above recommendations prior to our next meeting on March 23 in Aiken, S.C. Recommendations 75 and 76 are also being forwarded to John Hankinson of the Environmental Protection Agency and Lewis Shaw of the South Carolina Department of Health and Environmental Control.

We appreciate your thoughtful consideration of the enclosed advice.

Sincerely,

Ann Loadholt
Chairperson

cc: Tom Heenan, DOE-SR
Fred Butterfield, EM-22
Karol Hazard, EM 22
Chuck Rice, INEEL SSAB
John Hankinson, EPA-Region IV
Lewis Shaw, SCDHEC
SRS CAB Members
SSAB Chairs
NMED Hearing Clerk

Savannah River Site

CITIZENS ADVISORY BOARD

Recommendation 74
January 26, 1999

SRS Spent Nuclear Fuel Management Draft Environmental Impact Statement

For the past two years, the SRS CAB has had the highest interest in the development of a treatment option for Spent Nuclear Fuel (SNF) so that the material can be shipped to a federal repository on a defined schedule.

The CAB has reviewed the SRS Spent Nuclear Fuel Management draft EIS and has the following comments:

1. We acknowledge that the melt and dilute process has high expectations of technical success that will provide a waste form that will be acceptable for repository storage. | L6-1
2. We would be immediately concerned if DOE relied on the direct disposal technology as long as there continued to be significant uncertainty as to the waste form acceptability. | L6-2
3. The early expenditure of design and construction funds for the melt and dilute process should not undermine the funding for other established site missions. | L6-3
4. We note that as long as the canyons continue to operate for the at-risk SNF, as well as other current or not yet defined missions, they can be reexamined as a viable treatment option. No canyon shutdown decision should be made before the cost, time and technical uncertainties of the melt and dilute option have been resolved. | L6-4

The SRS CAB requests periodic updates on the progress of SNF management at SRS and recommends that DOE continue on its present course of action.

Response to comment L6-1: Comments noted.

Response to comment L6-2: See response to comment L5-3.

Response to comment L6-3: DOE requested funding for fiscal year 2000 to continue research and development activities and to begin facility design. The funding received for FY 2000 will allow design completion of the L-Area Experimental Facility (LEF). LEF is a pilot test facility that will demonstrate the feasibility of the melt dilute technology. LEF is scheduled to be constructed and placed online by the end of

FY 2002. Current funding for the melt and dilute technology does not undermine the funding for other established site missions.

Current funding for the melt and dilute technology does not undermine the funding for other established site missions.

Response to comment L6-4: Under the current planning basis DOE will have high confidence in the acceptability of the waste form before the canyons cease operations.

SPENT NUCLEAR FUEL MANAGEMENT EIS

The nonproliferation implications associated with the continued use of chemical separation process are minor compared with those associated with the surplus plutonium disposition program. The amount of Pu separated from all of the fuel would be small compared with the amount from the excess weapons. Nonproliferation applies to all Pu, regardless of its source. The safeguarding of Pu at SRS has been effective for many years.

L7-1

DOE should get serious about getting rid of all these small lots of spent fuel. As long as the F and H canyons are operable, the fuel should be chemically processed and the separated Pu should be stored with the Pu from the weapons. Repackaging, melting and compressing are no solutions to ridding the environment of the fission products and the Pu in the fuel. When the foreign fuel arrives, it should be put in the dissolver tank and processed. This cuts down on the storage requirements. As space is available, put the odds and ends of fuel in the dissolver. Small amounts of HEU can be put in the dissolver with lots of LEU fuel. Do a U235 analysis on the solution for criticality purposes. Proliferation requirements can be met by measuring the Pu in each tank of dissolver solution and the amount of separated Pu.

L7-2

L7-3

L7-4

L7-5

I understand that DOE has aluminum clad fuel at other DOE sites. I am sure that some of this fuel is in a hazardous condition. This fuel should be shipped to SRS and chemically processed.

L7-6

Let's clean up all of the spent fuel.

FROM: Robert F. Overman
724 Brucewood St.
Aiken SC 29801
Phone: (803)649-2782

Robert F. Overman

February 6, 1999

Response to comment L7-1: Comment noted.

Response to comment L7-2: DOE's proposed action is to select a new non-chemical processing technology that would put aluminum-based SNF into a form or container suitable for direct placement in a geologic repository. DOE's preferred alternative is to treat the majority of SNF using a non-separations-based technology, i.e., melt and dilute. DOE proposes to apply the Conventional Processing technology only to a relatively small volume of aluminum-based SNF at SRS (about 3 percent by volume and 40 percent by mass) before a new treatment facility is in place. The rationale for this processing is to avoid the possibility of future urgent actions, including expensive recovery actions that would entail unnecessary radiation exposure to workers, and in one case, to manage a unique waste form (i.e., core filter block).

Response to comment L7-3: Any plutonium recovered from SNF by the Conventional Processing technology would be considered surplus to the nation's nuclear weapons program and would be placed in storage at SRS pending disposition under decisions DOE will reach as part of the Surplus Plutonium Disposition EIS currently being prepared.

Response to comment L7-4: Regardless of the SNF management technology selected, includ-

ing processing, all of the SNF fission products and all of the plutonium in the SNF is planned for eventual placement in a geologic repository. In other words, the ultimate destination for all elements of SNF considered in this EIS is the same.

Response to comment L7-5: There is sufficient basin storage capacity for foreign fuel receipts. In addition, canyon capacity is fully utilized in the near-term to accomplish stabilization of other nuclear materials at the SRS (not considered in this EIS) and the material proposed for processing in this EIS. It is not cost-effective to maintain the canyons in operation over the time period needed to treat all the aluminum-based SNF expected to be received at the SRS through 2035.

Response to comment L7-6: In accordance with previous DOE decisions evaluated in the Programmatic SNF EIS (DOE/EIS-0203, 1995) domestic aluminum-clad SNF, except that at Hanford, will be consolidated at SRS until approximately 2035, and SRS will receive aluminum-based SNF from foreign research reactors until 2009.

See also the response to comment L7-2.

9469 Caughman Road
Columbia, SC 29209
phone: 803-783-6353
e-mail: wit111@aol.com

February 4, 1999

Andrew Grainger
NEPA Compliance Officer
U.S. Department of Energy
Savannah River Operations Office
Building 742A, Room 183
Aiken, SC 29802

Dear Mr. Grainger:

I am writing with regards to the draft Environmental Impact Statement for the Savannah River Site Spent Nuclear Fuel Management. I am a concerned citizen and a member of a number of organizations including SCOPE, Sierra Club, and Midwest Renewable Energy Association. I appreciate the opportunity to make comment on this EIS.

I'd like to make several recommendations for the final EIS. They are as follows:

1. In preparing the final EIS, DOE should make a long-term analysis of all issues. It seems that most past decisions regarding the nuclear industry have been extremely short-sighted. If the U.S. is to remain a world leader, it is necessary that its agencies look at the best interests of citizens far into the future, not just in the present generation. | L8-1
2. I believe DOE should make research into the melt and dilute technology a high priority. It is critical that DOE determines whether the facility off-gas system can safely capture airborne radionuclides. In addition, DOE should make the findings of such research publicly available for review. DOE should also utilize an independent technical review board to evaluate engineering designs. | L8-2
| L8-3
3. DOE should emphasize the importance of past decisions to phase out reprocessing. These decisions were made for sound reasons. With this in mind, during the phase out period, only materials which demonstrate an immediate hazard should be reprocessed. As an agency of the U.S. people, DOE should take a leadership position in minimizing the spread of nuclear weaponry. As such, DOE should set an example by showing the world that nuclear materials can be safely managed without separating weapons usable materials. | L8-4
| L8-5

4. When preparing the final EIS, DOE should be more detailed. Quantitative and qualitative data should be thorough and accurate enough to allow independent validation of the assumptions and conclusions which are presented in the report. This is essential to avoid providing distorted information which will make reprocessing appear to be a more favorable decision than it actually is. | L8-6
5. In a similar vein, when preparing this report, DOE should take a comprehensive look at all related issues, particularly all potential reprocessing issues in the near and distant future. Any decisions should assure that cumulative, rather than discrete impacts are considered. The public should be made aware of future reprocessing activities, the point at which reprocessing facilities will be safely decommissioned, and the manner in which nuclear materials will be managed in the future. | L8-7
| L8-8

Again, I appreciate the opportunity to give my feedback on this report. Thank you for taking my concerns into consideration. Please bear in mind, this is an ideal opportunity for the U.S. to pursue new and safer methods for managing nuclear materials, rather than continually managing nuclear materials in a manner which has proven to be deleterious to the U.S. and world citizenry.

Sincerely,

Christine Witkowski

Christine Witkowski

Response to comment L8-1: In this EIS, DOE has analyzed the treatment and interim storage of SNF at SRS through the year 2035. Beyond 2035, it becomes increasingly difficult to reasonably anticipate program needs. For planning purposes, DOE plans to begin transferring treated SNF to a geologic repository by 2015. Impacts from utilizing a geologic repository for commercial SNF and DOE SNF and high-level waste are currently being evaluated under the Yucca Mountain Repository EIS.

Response to comment L8-2: As described in the response to comment L2-18, DOE has been performing research for two years on the melt and dilute process with surrogate materials. DOE also is embarking on a pilot program (L-Area Experimental Facility) to test the melt and dilute process on irradiated fuel. These tests have not identified any issues that would lead DOE to believe that the melt and dilute process could not be successfully implemented. The design and construction of a full-scale facility would need to be developed in the context of constrained, out-year budget targets, and funding for such a facility would need to be balanced against other priorities at SRS. DOE is in the process of developing a schedule that can be used as a baseline for near-term planning and budgeting purposes. The FY 2000 budget has been established and includes funding for design completion of the L-Area Experimental Facility (LEF). LEF is scheduled to be constructed and placed online by the end of FY 2002.

Response to comment L8-3: As described in response to comment L2-17, DOE's objective is to develop an offgas system that will ensure the safety of workers, the public, and the environment; comply with DOE Orders and Standards; and meet all regulatory requirements.

And, as described in response to L2-20, the development and review process is, in general, open to interested people from throughout the DOE system. Briefings on the technology are conducted regularly with the Office of Radioactive Waste Management, and also with DOE and contractor experts from throughout the DOE complex in the area of management of spent

nuclear fuel. Briefings also are conducted for the Defense Nuclear Facilities Safety Board and their staff. The Nuclear Regulatory Commission and their contractors review project documentation and provide DOE feedback on technical issues. In areas where technical uncertainty still remain, such as the design for the offgas system, DOE will initiate peer review from appropriate technical experts from the DOE complex exclusive of Savannah River Site. For members of the public, DOE will make available, upon request, select summary reports such as the report on off-gas system development to be issued in May 1999. Contact Randy Ponik, DOE-SR (803) 952-2549 or Randall.Ponik@SRS.gov to request additional technical information on technology development activities.

Response to comment L8-4: DOE's proposed action is to select a new non-chemical processing technology that would put SNF into a form or container suitable for direct placement in a geologic repository. DOE's preferred alternative is to treat the majority of SNF using a non-separations-based technology, i.e., melt and dilute. DOE proposes to apply the Conventional Processing technology only to a relatively small volume of SNF at SRS (about 3 percent by volume and 40 percent by mass) before a new treatment facility is in place. The rationale for this processing is to avoid the possibility of future urgent actions, including expensive recovery actions that would entail unnecessary radiation exposure to workers, and in one case, to manage a unique waste form (i.e., core filter block).

| EC

| EC

Response to comment L8-5: One of the primary elements of the proposed action of this EIS regarding the management of aluminum-based SNF is to select a non-chemical separations based SNF treatment technology.

Response to comment L8-6: DOE strives to make all data and information available so that members of the public and other interested parties can independently assess the methodology, assumptions, and conclusions presented in the EIS. All reports and technical documents used

to support the analysis in the EIS are available for review at the DOE public reading rooms at the Gregg-Graniteville Library of the University of South Carolina, Aiken, South Carolina, and in DOE Headquarters in the Forrestal Building, 1000 Independence Ave., Washington, DC.

Response to comment L8-7 and L8-8: All the materials that DOE has decided to process or is proposing to process are in the current planning basis. DOE is evaluating disposition alternatives for other materials which might be potential candidates for canyon processing. Decisions on these materials would be made only after appropriate NEPA review. The impacts of these potential materials have been included in the cumulative analysis for this EIS.

Patterson, Karen

From: Oliver, James
Sent: Friday, February 12, 1999 4:23 PM
To: Phil Young; Bruce Bradford; Karen Patterson
Subject: FW: draft EIS for spent nuclear fuel

Subject: draft EIS for spent nuclear fuel
Author: Wit111@aol.com at Mailhub
Date: 2/12/99 1:27 PM

Dear Mr. Grainger,

I attended the Columbia hearing and mailed in comments earlier, but I have several additional comments to add.

1. Since the spent nuclear fuel is likely to be with us for a very long time, and the options available at present are not ideal, why not wait a year or two until the technological advancements offer us additional options for dealing with this spent nuclear fuel. What is the rush? | L9-1
2. I think the Department of Energy has a responsibility to do a better job of educating the public about nuclear issues, beginning with very simple terms and advancing to more technical explanations, so that people can be informed participants in the decision-making process. The mass media has been used in the past to great advantage in educating the public about important issues. Since nuclear issues affect the public health, economics, and environment, it is the government and its agencies' responsibility to help the public make wise choices, instead of making all of the choices for them. | L9-2
| L9-3
3. Similarly, I think the DOE has a responsibility to do a better job of publicizing any and all public hearings. It may seem futile to you, but it is part of your duty according to the regulations. Additionally, I think the DOE has a responsibility to make the public hearings less jargon oriented, using more layperson's terminology. Also, it should be much clearer when the time for questions is, versus when the time for public comments is. I got the impression at the recent hearing in Columbia that the period for public comment was very neglected. | L9-4
| L9-5

Thank you for taking these points into consideration.

Sincerely,

Christine Witkowski
Concerned Citizen and member of Southern Coalition Opposing Plutonium
Energy (SCOPE)
9469 Caughman Road
Columbia, SC 29209
803.783.6353
wit111@aol.com

Response to comment L9-1: The options (i.e., alternatives) evaluated in this EIS covered all feasible and currently reliable alternatives. DOE does not expect any new SNF treatment technologies to be presented in the near future. Regardless of which technology is selected, implementing the process will require several years. DOE is committed to ensuring that the health and safety vulnerabilities the fuels could pose in the future if they remain in the wet storage basins are eliminated. See Sections 2.2 and 2.6.1.

Response to comment L9-2: DOE is continually exploring ways to better inform and educate the public about the management of nuclear materials so that they can better participate in the decisionmaking process. For example, in 1994, DOE established an Environmental Management Site Specific Advisory Board at SRS. This board is composed of members of the general public. Regulatory agencies are *ex officio* members of the Board. Through this stakeholder organization, DOE provides to the public information and education on environmental management activities at SRS. All Citizen Advisory Board meetings are advertised in local media and are open to the public. Meetings are held on a regular basis in Aiken, Augusta, Savannah, Hilton Head, and Charleston. The public is encouraged to participate in the activities of the Board.

Response to comment L9-3: DOE is committed to making informed decisions utilizing input from the public. One of the primary ways in which the Department solicits input from the public is through the NEPA process which emphasizes public involvement as a key to providing information to DOE regarding major activities that could have a significant impact on the environment. Public input is solicited at several points in the NEPA process. DOE also maintains a mailing list of interested organizations and members of the public, and mails copies of the Notices, EISs, and Records of Decision to people who have expressed an interest in the activity or Site. DOE uses this extensive public involvement process to collect input from the public that it then considers when making decisions described in the EISs.

DOE welcomes any specific suggestions for more effectively notifying interested parties of public meetings.

TC

Response to comments L9-4 and L9-5: Comment noted.



DEPARTMENT OF HEALTH & HUMAN SERVICES

Public Health Service

Centers for Disease Control
and Prevention (CDC)
Atlanta GA 30341-3724

January 19, 1999

Andrew R. Grainger, NEPA Compliance Officer
Savannah River Site
Building 742-A, Room 185
Aiken, SC 29802

Dear Mr. Grainger:

We have completed our review of the Draft Environmental Impact Statement (DEIS) for Spent Nuclear Fuel Management (DOE/EIS-0279D). We are responding on behalf of the U.S. Public Health Service, Department of Health and Human Services. Technical assistance for this review was provided by Mr. C.M. Wood, Radiation Studies Branch, National Center for Environmental Health, Centers for Disease Control & Prevention.

Generally, we believe this DEIS addresses our potential concerns, and we have no major comments to offer at this time. However, we do offer the following editorial comments for your consideration:

SUMMARY

Page S-3, para S.2.3: the last paragraph defines "metric tons heavy metal" twice in the same paragraph.

L10-1

The Summary contains a list of the alternative technologies for processing fuel (Melt and Dilute, Mechanical Dilution, etc), with a brief discussion of the pros and cons of each. We recommend a brief description of each of the processes at the beginning of each discussion.

L10-2

DEIS

Page 2-11 "Neutron poison material *could* be added if necessary." Change to "*will* be added." (If they follow the weight restrictions shown on page 2-3 in order to avoid a critical configuration, why would it be necessary to add poison?)

L10-3

Thank you for the opportunity to review and comment on this DEIS. Please send us a copy of the Final EIS.

Sincerely,

Kenneth W. Holt

Kenneth W. Holt, MSEH
Special Programs Group (F16)
National Center for Environmental Health

Response to comment L10-1: The second definition will be removed.

Response to comment L10-2: A brief description of each alternative technology has been added to the Summary.

Response to comment L10-3: This is correct as written in the EIS. The low enrichment of the fuel varies. Because some of the fuel is low-enriched, it may not be necessary to add a neutron poison to these fuels to prevent a criticality.



UNITED STATES DEPARTMENT OF COMMERCE
National Oceanic and Atmospheric Administration
NATIONAL MARINE FISHERIES SERVICE

Southeast Regional Office
9721 Executive Center Drive N.
St. Petersburg, Florida 33702

January 11, 1999

Mr. Andrew R. Grainger
NEPA Compliance Officer
Savannah River Operations Office
Building 742-A, Rm. 185
Aiken, South Carolina 29802

Dear Mr. Grainger:

The National Marine Fisheries Service (NMFS) has reviewed the Draft Environmental Impact Statement (DEIS), for Savannah River Site (SRS) Spent Nuclear Fuel Management (DOE/EIS-0279D). The DEIS addresses action for the safe and efficient management of spent nuclear fuel and targets stored at Department of Energy's SRS facility located near Aiken, South Carolina. The DEIS was transmitted to us by your letter of December 22, 1998.

According to the DEIS, the proposed action and alternatives would utilize existing uplands, includes no plans to excavate or fill waters of the Savannah River and its tributaries, and would comply with Federal and state requirements for containment and discharge of surface water and wastes. Based on these considerations, we concur with your determination that impacts to aquatic resources should be negligible and are sufficiently addressed in the DEIS.

L11-1

As noted in Section 7.1.2 of the document, consultation with the NMFS is required in connection with Section 7 of the Endangered Species Act of 1973, as amended (ESA). According to your Distribution List, a copy of the DEIS was sent to Mr. Charles Oravetz of our Protected Resources Division, for review. Completed coordination with that office is required to fulfill your ESA obligations.

L11-2

Mr. David Rackley at our Charleston Area Office is available to address any questions or comments that you may have in connection with our review of the DEIS or the proposed action as it relates to the Savannah River and its aquatic resources. He may be reached at 219 Fort Johnson Road, Charleston, South Carolina 29412-9110, or at (843) 762-8574.

Sincerely,

Andreas Mager, Jr.
Assistant Regional Administrator
Habitat Conservation Division



cc:

F/SER45

F/SER3

PSP - Fruchter

Response to comment L11-1: Comment noted.

Response to comment L11-2: DOE contacted the National Marine Fisheries Service (NMFS) on December 21, 1998, requesting comments on the Draft Spent Nuclear Fuel Management EIS, especially in regards to their responsibilities under the Endangered Species Act (ESA). The action area for the proposed action and alternatives is adjacent to the Savannah River which has one of 20 distinct population segments of the shortnose sturgeon (*Acipenser brevirostrum*), listed as endangered under ESA.

On June 22, 1999, DOE Savannah River Operations Office received a formal reply (see Letter L-21). Based on the facts presented in the Draft EIS, "...the NMFS does not believe that the proposed action and alternatives would affect the shortnose sturgeon nor any another species protected by the ESA under NMFS purview. This concludes the DOE's consultation responsibilities under Section 7 of the ESA for the proposed action and alternatives described in the December 1998 DEIS for species under NMFS purview."

Mr. Andrew R. Grainger:

We were unable to attend the meeting in Columbia, South Carolina on January 28th and wanted to express our opinion. The processing of spent nuclear fuel must be done, we agree.

1) But the descriptions given are not written for the public to understand; they should be rewritten in a non-technical manner.

| L12-1

2) Also, no matter what method is used to process spent nuclear fuel, there will be the obvious contamination of clothing and materials. Currently those materials travel on public roads and come into our quiet, residential neighborhood, to the INS nuclear laundry. PLEASE change this policy and return the nuclear laundry facilities to being on-site at the Savannah River Site! If we must contaminate certain areas, let's keep it in a contained area and not within a mile or two radius of where we live, work, shop, and exercise. The INS laundry does not need to be in our otherwise clean and beautiful neighborhood.

| L12-2

Thank you for your consideration.

Sincerely,
Stan and Elaine Frick
2921 Ocoola Street
Columbia, SC 29205
(803) 799-6437

Response to comment L12-1: The management of spent nuclear fuel is a very technical process. DOE presented the material in the EIS as simply as possible, but some of the technical discussions do not lend themselves to extreme simplification. The summary is a non-technical description of the process. The commenter may find its descriptions more understandable.

Response to comment L12-2: The operation of a nuclear laundry facility is out of the scope of this EIS. While each of the alternatives addressed in the spent nuclear fuel EIS (including No Action) would generate laundry that would be sent to an appropriately licensed and permitted facility, the small increase in volume would not affect the projected impacts.

TC



University of Pittsburgh
at Johnstown

Johnstown, Pennsylvania 15804
Telephone 814-269-7000

6 February 1999

Andrew R. Granger
NEPA Compliance Officer
U.S. Department of Energy
Savannah River Operations Office
Building 742A, Room 183
Aiken, SC, 29802

Attention : Spent Nuclear Fuel Management EIS

Dear NEPA Compliance Officer:

Enclosed are my comments on the Draft Environmental Impact Statement, Spent Nuclear Fuel Management, Savannah River Site, DOE/EIS-0279D. Please note that the opinions presented here do not necessarily reflect the position of the University of Pittsburgh at Johnstown.

I hope that the issues raised herein will be addressed in the final EIS.

Sincerely,

William A. Lochstet

William A. Lochstet
voice: 814-269-2937
fax: 814-269-2022

Transforming the Present—Discovering the Future

Some Suggestions on
Spent Nuclear Fuel Management
at SR5

by

Wm. A. Lochstet
The University of Pittsburgh
at Johnstown*
February 1999

The Department of Energy (DOE) has prepared a Draft Environmental Impact Statement on the Savannah River Site Spent Nuclear Fuel Management at the Savannah River Site, DOE/EIS-0279D. This document presents a useful catalog of the spent nuclear fuel (SNF) in appendix C, along with an estimate of the radionuclide inventories as table C-7. This table is very useful in recognizing the magnitude of the problem. DOE anticipates that it will prepare this material for a geologic repository, or other off-site disposition.

DOE has classified the SNF into five groups (labeled A-F). Group A contains fuels from Experimental Breeder Reactor-II, and other sources. This group contains about 20 metric tons of heavy metal, and is slated to be reprocessed in the F or H canyon facilities. The bulk of the material in groups B, C, and D are aluminum based fuel, with aluminum cladding. Failed fuel and loose oxide in these groups are listed for reprocessing in F or H canyon. The major part of the group B,C,D fuel, containing about 28 metric tons of heavy metal, is identified for a "melt and dilute" process. The material in groups E and F are expected to be transferred off site, and not processed by "melt and dilute."

The "melt and dilute" process involves placing the fuel elements in an induction heated melter with additional aluminum and depleted uranium as needed. After melting, would be cast into a form about 16 inches in diameter, and up to 33 inches long. It is noted that melting is a fuel failure mode and would result in a release of some fission products. The resulting ingots would be sealed into stainless steel containers before shipment to a geologic repository.

At present, the repository acceptance criteria do not exist. The end product of the "melt and dilute" process is contained in a stainless steel canister that would certainly be breached by geological pressure. The aluminum and heavy metal ingot would enter into geochemical reactions. This is not a stable waste form, and must not be considered for a geologic repository.

L13-1

* Affiliation for identification purposes only.

Page 2
SRS-SNF

Although DOE does identify three possible processes that would result in a glassy end product, it does not endorse any of them. Such a choice would produce a wasteform more in keeping with the requirements of a geologic repository. Recently, DOE has decided on a ceramic form, synroc, for plutonium immobilization. A recent comparison of glass and synroc by Allison Macfarlane (Science & Global Security, Vol 7, No 3, 1998, pp 271-309) also concludes that synroc is preferable. It is then clear that DOE should modify its decision and convert this material to synroc, rather than "melt and dilute."

L13-2

L13-3

Response to comment L13-1: DOE-SR is working closely with the Nuclear Regulatory Commission (the federal agency that would license the operation of a geologic repository) to ensure that the final product from the selected SNF treatment technology would be acceptable for disposition. DOE expects repository waste acceptance criteria to be established well before the melt and dilute system begins operation. The EIS has been revised to discuss in greater detail the expected repository acceptance criteria and compare the treatment technology products to that criteria. This information is discussed in Sections 2.2.1.

See also response to comment L5-3 for additional information.

Response to comment L13-2: DOE is doing research and development to ensure that the final product of melt and dilute technology will meet repository waste acceptance criteria.

Response to comment L13-3: One of the primary reasons DOE has selected Melt and Dilute as the preferred technology to process aluminum-based SNF is because DOE is confident that the waste form produced by this technology will meet the geologic repository waste acceptance criteria. It also can be implemented relatively easily and in a timely fashion. DOE has no data to suggest synroc would be preferable to the Melt and Dilute process product.

Oliver, James
From: NEPA@srs.gov
Sent: Tuesday, February 09, 1999 8:24 AM
To: OliverJ@ttnus.com; Karl.Waltzer@srs.gov; Drew.Grainger@srs.gov
Subject: Comments on Draft Spent Nuclear Fuel Management EIS

Comments.

Forward Header

Subject: Comments on Draft Spent Nuclear Fuel Management EIS
Author: goergen@groupz.net at Mailhub
Date: 2/9/99 12:09 AM

Dear Mr. Grainger,

Please find attached my written comments to supplement those given in the Public Meeting on February 2, 1999 in North Augusta. If you have any questions you may contact me at 849-4097.

Sincerely yours,
Charles R. Goergen


Spent_Fuel_EIS.doc

February 8, 1999

Savannah River Site Spent Nuclear Fuel Management EIS
Charles R. Goergen
510 Boardman Road
Aiken, SC 29803

I am providing this comment on the Draft of the Spent Nuclear Fuel Management EIS and its supporting documents. I support a decision that takes advantage of currently operating facilities to process the maximum amount of spent, aluminum based fuel and incorporates an alternative strategy when the inventory is substantially reduced and only minimal amounts of new receipts are forecasted. Any decrease in future receipts would further delay the onset of requirements for an alternative technology. In times of dwindling budget, the Department of Energy should maximize use of current investment and delay construction of facilities until absolutely necessary.

L14-1

The Nonproliferation Study has several assumption faults which makes the outcome conclusion flawed. The authors state that Conventional Reprocessing/Chemical Separation in section 3.4 "the plutonium would be placed in storage and dispositioned with other surplus weapons-usable plutonium". For the 97% of the spent fuel proposed to be dispositioned by alternative technology, when processed through H-Canyon, the plutonium is never separated from the fission products and follows the fission product waste stream where it would be processed through the DWPF into glass. It is not a proliferation concern.

L14-2

The Cost Study neglects the credit offset for sale of the HEU diluted to LEU and use in TVA power reactors. This would allow the US Government and taxpayers to recover the Separative Work Units (SWUs) already invested in the HEU, whether the DOE or TVA receives the share of savings. I contend that would make conventional processing the maximum amount of material using current, well characterized processes a clear, cost-effective winner. With the majority of the material chemically processed, the size of an alternative technology facility could be much smaller and not needed until 2015 or later. This would stretch out the funding, research, and construction over many years. Furthermore, the H-Canyon will not be able to shut down in an immediate manner. Other missions such as neptunium stabilization will allow for concurrent use and shared costs thereby reducing the total cost for incremental fuel processing. Processing should continue up to the logical point at which decontamination and decommissioning activities encompass the HEU fuel process operations.

L14-3

L14-4

L14-5

I urge the Department to heed the NAS Task Team option on page 7 of the Research Reactor Aluminum Spent Fuel, Treatment Options for Disposal. "DOE should have given more careful consideration to the conventional reprocessing option for treating aluminum spent fuel. There appear to be several technical advantages to this option over the others considered by the Task Team. This treatment option has been demonstrated to work for aluminum spent fuel from production reactors, the costs and risks are well known, the necessary facilities are currently in operation at Savannah River, and the

L14-6

waste form (borosilicate glass) will likely be acceptable for disposal at the repository. Reprocessing even a portion of the aluminum spent fuel could significantly reduce the overall costs of treating the total aluminum spent fuel inventory by alleviating the need for additional spent storage space at Savannah River and eliminating the problems with odd-sized fuel elements that may be difficult to process by other methods." Also on pages 15, 16, and 17, "The concern with conventional processing appears to be mainly one of policy and is related to the use of reprocessing for waste management generally rather than any specific concern about reprocessing this particular fuel type." "It is the P.I.'s opinion that the acceptability of conventional reprocessing might be increased if it were redesigned as a reprocess-and-dilute operation, in which spent fuel is conventionally processed and the separated ^{235}U is diluted with ^{238}U to produce low-enriched uranium before it leaves the reprocessing facility." This dilution to non-weapons useable LEU is the current option for spent SRS aluminum fuel elements and is underway. Let's take full advantage of the country's current capabilities to handle the majority of the material in question.

Response to comment L14-1: DOE's preference is to pursue non-chemical separations based technologies to manage the aluminum based SNF at SRS. DOE intends to utilize the canyons to process SNF that may present a potential health and safety vulnerability.

Response to comment L14-2: The statement in Section 3.4 of the Nonproliferation Study on the separation of plutonium pertains to fuels designated for F-Canyon.

Response to comment L14-3: The cost study referenced in the EIS states that uranium credits in the range of \$110 million to \$150 million were considered in developing the cost report. However, due to the recently signed agreement between Russia and the United States that calls for the potential deferment of HEU sales, no actual value was assigned in the report for calculating life-cycle costs associated with processing activities. Section 2.6.5 of the EIS has been revised to clarify this point.

Response to comment L14-4: The cost report referenced (and summarized) in the EIS evaluated the cost for processing the aluminum-based SNF to be managed at SRS. The processing cost estimate was made using assumptions similar to those described by the commenter, i.e., processing in the canyons would occur until SNF receipts were low and then a smaller-capacity treatment facility would be used through 2035. Section 2.6.5 of the EIS has been revised to clarify this information.

Response to comment L14-5: See response to comment L14-1.

Response to comment L14-6: See response to comment L14-1. DOE fully evaluated the alternative of using the existing chemical separations facilities at SRS to manage aluminum-based SNF at SRS.

Environmentalists, Inc.
1339 Sinkler Road
Columbia, SC 29206
(803) 783-3000

February 17, 1999

Andrew P. Grainger
NEPA Compliance Officer
Savannah River Site
Building 742-A, Room 185
Aiken, SC 29802

Dear Mr. Grainger:

Re: Draft Environmental Impact Statement (DEIS) related to Spent
Nuclear Fuel Management at the Savannah River Site (SRS).

Please add the following comments to those that I submitted on behalf of
Environmentalists, Inc. (E.I.) in a letter of February 7, 1999:

1. It is unclear why the Department of Energy (DOE) failed to identify and
use the following studies when preparing its DEIS:

- National Academy of Science – National Research Council, a
report to the Division of Reactor Development and Technology,
U.S. Atomic Energy Commission, Committee on Geologic
Aspects of Radioactive Waste Disposal, May 1966. L15-1
- Geological Survey Water Supply Paper of the U.S. Department
of the Interior, Geology and Ground Water of the Savannah River
Plant and Vicinity, South Carolina, by George E. Simple,
prepared in cooperation with the Atomic Energy Commission,
1967. L15-2
- U.S. Atomic Energy Commission's Safety Evaluation in the
matter of Allied-Gulf Nuclear Services' Barnwell Nuclear Plant,
Docket 50-332, October 9, 1970, Geologic and Hydrologic
Review by U.S. Department of the Interior, pages 108-113. L15-3

2. It is unclear why transcripts of the Allied-General Nuclear Services
proceedings related to the Barnwell Nuclear Fuel (reprocessing) Plant
were not identified and used as references for the DEIS. L15-4

- | | |
|---|-------|
| 3. There is a lack of information in the DEIS regarding reprocessing experience in this country (Nuclear Fuel Service in West Valley, NY) and in France and England. | L15-5 |
| 4. Why is reprocessing being considered by DOE when Canyons F and H are expensive to operate because of their size and age? | L15-6 |
| 5. Reprocessing plants in the Hanford, Washington and Idaho are no longer operating in accordance with the decision in 1992 of the Bush Administration to phase out reprocessing. Why hasn't reprocessing been phased out at SRS? | L15-7 |
| 6. The DOE failed to present adequate information and evidence regarding the agency's selecting the melt-and-dilate option as its preferred alternative for managing most of the aluminum based spent nuclear fuel. | L15-8 |

Recommendations

- | | |
|---|--------|
| 1. That the DOE use footnoting in its Final Environmental Impact Statement (FEIS) which identifies the title of reference documents, as well as including author(s) and page on which supporting evidence appears. | L15-9 |
| 2. That the DOE prepare a FEIS which corrects the deficiencies of the DEIS and therefore fulfills to the "fullest extent possible" the substantive and procedural duties as required by the National Environmental Policy Act (NEPA). | L15-10 |

Thank you for your attention to the comments, questions, and recommendations of Environmentalists, Inc.

Sincerely,



Ruth Thomas, President
Environmentalists, Inc.
1339 Sinkler Road
Columbia, SC 29206
(803) 782-3000

Response to comment L15-1: The geologic aspects of radioactive waste disposal are outside the scope of this EIS and are being addressed in other NEPA analyses and related studies.

Response to comment L15-2: DOE makes an effort to describe the affected environment on the basis of the most current information available. More recent studies that were used to prepare the EIS incorporate all relevant information from these earlier documents.

Response to comments L15-3 and L15-4: The proceedings mentioned by the commenter were from the early 1970s. DOE prefers to use the most recent scientifically validated data available. In addition, Barnwell Nuclear Fuel Plant never operated and did not process nuclear material. DOE prefers to use data from operating facilities. The extensive database of SRS operating experience has been used in this EIS to assess the expected impacts of the operation of existing facilities, and this same database has been applied to estimate the potential impacts of operation of proposed spent nuclear fuel management facilities.

Response to comment L15-5: The SRS Spent Nuclear Fuel Management EIS focuses on predicting the environmental impacts of DOE's alternatives for SNF management, including processing. To that end, DOE has used actual SRS processing facility emissions information to predict the emissions of future operations associated with SRS processing facilities.

Response to comment L15-6: Processing in F and H Canyons is considered a reasonable alternative because (1) DOE has more than 40 years of experience managing spent nuclear fuel by processing in SRS canyons, (2) processing would produce a waste form that has a very high probability of meeting the repository waste acceptance criteria, and (3) other alternatives are also expensive to build and operate.

Response to comment L15-7: Chemical separations operations are occurring at SRS because DOE is using the canyons to stabilize nuclear materials that represent potential health and safety risks in their current storage configuration. The decisions to use processing capabilities have been documented in a number of Records of Decision, including those following the F-Canyon Plutonium Solutions EIS, the Interim Management of Nuclear Materials EIS, and the Rocky Flats Plutonium Residues EIS. These decisions are consistent with DOE's Implementation Plan for Defense Nuclear Facilities Safety Board Recommendation 94-1, wherein the Board recommended that DOE take steps, including use of the processing facilities, to stabilize nuclear materials that represented health and safety risks. Section 1.6 of this EIS provides information on NEPA documents and decisions relevant to canyon operations.

Response to comment L15-8: DOE prefers the melt and dilute technology because the technology is non-chemical separations based, it provides significant waste reduction in comparison to conventional processing, it is fully compatible with United States nonproliferation objectives, and is among the simplest of the new treatment technologies to implement. This information is provided in Section 2.4.3.1 of the EIS.

Response to comment L15-9: References are provided at the end of each chapter as per accepted NEPA practice.

Response to comment L15-10: This Final EIS corrects inconsistencies and clarifies sections that DOE determined, after review of public comments or internal reviews, required modification. This EIS is based on the best and most current information available, fully describes DOE's proposal and alternatives, and evidences careful consideration of all comments received.

| EC

Comment Letter L-16

**Nuclear Control Institute
Natural Resources Defense Council
Institute for Science and International Security**

June 24, 1999

NUCLEAR CONTROL INSTITUTE (NCI)
NATURAL RESOURCES DEFENSE COUNCIL (NRDC)
INSTITUTE FOR SCIENCE AND INTERNATIONAL SECURITY (ISIS)

June 24, 1999

Secretary Bill Richardson
Department of Energy
1000 Independence Avenue, S.W.
Washington, D.C. 20585

Dear Secretary Richardson:

We are writing to express our opposition in the strongest terms to the suspension and possible elimination of the "alternative technology" program for highly enriched-uranium spent nuclear fuel at the Savannah River Site. It is imperative that you take immediate steps to ensure the continuance of this program, as stipulated in the 1996 Record of Decision ("1996 ROD") for the Foreign Research Reactor Spent Nuclear Fuel Environmental Impact Statement.

The suspension of the alternative technology program is fundamentally inconsistent with the Department of Energy's current plans under the National Environmental Policy Act to use "melt and dilute," the preferred alternative technology, to treat aluminum-clad spent nuclear fuel for long-term disposal. The suspension of the program is also contrary to assurances the Department of Energy ("DOE") gave to our organizations when the disposition of U.S.-origin foreign spent fuel was being debated -- namely, that alternatives to reprocessing would be developed and implemented. Elimination of support for melt and dilute would leave DOE with no alternative for treatment except reprocessing in the SRS "canyons," which are extremely expensive to operate, create great volumes of liquid waste, and produce weapons-usable nuclear materials.

During the mid-1990s, DOE prepared an Environmental Impact Statement on disposal of highly enriched spent nuclear fuel generated by foreign countries but of United States' origin. This process was integral to DOE's Reduced Enrichment in Research and Test Reactors ("RERTR") program, a non-proliferation program designed to curtail the spread of weapons-grade uranium. Our organizations have long supported, and continue to support the RERTR program and to applaud its successes.

The 1996 ROD on the Foreign Research Reactor EIS contains two central elements: (1) U.S.-origin spent nuclear fuel would be returned to DOE for disposal; and (2) a program would be implemented by DOE to develop non-reprocessing technologies for managing spent fuel. Our organizations were instrumental in calling for investigation of non-reprocessing methods for managing spent nuclear fuel on non-proliferation, environmental, and economic grounds.

At the time the 1996 ROD was issued, and since, DOE assured our organizations that alternative technologies would be aggressively pursued. The progress of these efforts was affirmed in the Draft Environmental Impact Statement ("Draft EIS") on Spent Nuclear Fuel Management, published in December 1998. In the Draft EIS, DOE stated that it "has greater confidence now than at the time of the Foreign Research Reactor EIS's Record of Decision in the feasibility and availability of non-reprocessing technologies." Further, the Draft EIS selected one of the program's technologies, "melt and dilute," as the preferred option for processing both foreign and domestic aluminum-clad spent nuclear fuel.

Nevertheless, despite the 1996 ROD and the findings in the Draft EIS, over the past eight months DOE has removed 60% of the FY 1999 budget of \$10 million appropriated for the alternative technology program -- 40% in November 1998 and another 20% in April 1999. This money was apparently transferred to the F&H Canyon Exhaust Upgrades project. As a result of these transfers, in May 1999 DOE suspended the alternative technology program and reassigned all of its staff because the program's budget had been expended. The very existence of the alternative technology program thus now appears to be threatened.

DOE's actions violate the requirement of the 1996 ROD that alternative technologies be "aggressively pursued," at minimum, "through approximately the end of 1999." DOE's failure to comply with the basic terms of its 1996 ROD is without rational justification, is arbitrary and capricious and in violation of the Administrative Procedure Act. We are especially indignant because DOE's sudden reversal constitutes a breach of faith with those of us who worked closely with DOE in support of the RERTR spent fuel take-back program, a program we continue actively to support.

Furthermore, although DOE has submitted a reprogramming request to Congress to meet the added costs of the F&H Canyon Exhaust Upgrades, this canyon project currently holds in reserve about \$2 million of the funds transferred from the alternative technology program. These funds are needed to ensure continuation of the alternative technology program, but DOE nonetheless refuses to transfer them back.

DOE's position is completely contradictory: at the same time DOE determined that melt-and-dilute was the preferred alternative under the Draft EIS, DOE began defunding the alternative technology program. DOE's failure to safeguard the funding of the alternative technology program, as appropriated by Congress, is therefore indefensible.

We urge you to take immediate action to see to it that funding for the alternative technology program is restored and that the terms of the 1996 ROD are fulfilled. We further ask that you investigate, on an expedited basis, all aspects of the suspension of the melt and dilute program, which according to DOE provides substantial environmental benefits and is one of the "most proven" alternative technologies available. Time is of the essence. The disruption and uncertain future of the alternative technology program could lead its staff to choose not to return; similarly, the longer the suspension of work,

the more difficult it will be to reassemble the alternative technology team. Accordingly, your most important objectives should be to reactivate the program immediately and to ensure that funding is stable for the remainder of FY 1999 and in subsequent years.

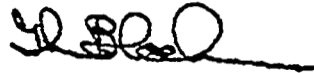
We request that you provide to us an explanation of how and why the funds were transferred from the alternative technology program, how this program was permitted to run out of funds, and what steps DOE intends to take to reactivate it. Further, we believe that it is essential that a meeting be scheduled with you and your staff as soon as feasible to discuss the status of the alternative technology program and DOE's plans to comply fully with the terms of the 1996 ROD and the pending EIS and Record of Decision on Spent Nuclear Fuel Management.

Thank you for your consideration of this urgent matter. We look forward to hearing from you shortly.

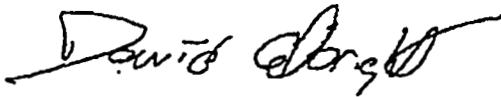
Sincerely,



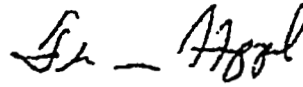
Tom Clements
Executive Director
Nuclear Control Institute



Tom Cochran
Director, Nuclear Program
Wade Greene Chair for Nuclear Policy
Natural Resources Defense Council



David Albright
President
Institute for Science and
International Security



Frank von Hippel
Professor of Public and International Affairs
Princeton University
(as an individual, not for his organization)

cc: Deputy Secretary T. J. Glauthier
Under Secretary Ernest Moniz
General Counsel Mary Anne Sullivan
Acting Deputy Assistant Secretary David Huizenga
Greg Rudy, Manager, Savannah River Operations

DOE Response to Comment Letter L-16

**Department of Energy**

Washington, DC 20585

September 24, 1999

Mr. Thomas Clements
Executive Director
Nuclear Control Institute
1000 Connecticut Avenue, NW
Suite 804
Washington, D.C. 20036

Dear Mr. Clements:

I have been asked to respond to your June 24, 1999 letter to the Secretary of Energy. In that letter, you raised the concern that the Alternate Technology Program at the Savannah River Site (SRS) had been suspended or possibly canceled altogether, and you requested the immediate resumption of all program activities. As you already may be aware, funding for the program was restored in June 1999. Program activities are presently underway, and further activities are planned for the future.

Because of certain requirements in the final Environmental Management FY 1999 budget authorization for SRS, and because of urgent emergent needs that arose during FY 1999, budgetary cuts in some planned programs, including the Alternate Technology Program, had to be made. While the alternate technology FY 1999 funding was cut to \$4 million, a higher spending rate was maintained in anticipation of achieving greater efficiency in other work or identifying additional funding sources. The \$4 million was completely expended by May 1999, and the technology development team was temporarily reassigned until additional funding could be identified. The additional funding has now been made available through an internal site reprogramming. The funding primarily came from the financial closure of recently completed line item projects. Two million dollars was restored to the program for the remainder of FY-1999. The reassignment of personnel was never considered a permanent action, and was not related to extending canyon operation.

The Department is in the process of preparing a final environmental impact statement (EIS) that considers the potential environmental impacts of alternative means of managing spent nuclear fuel at SRS, including those technologies that are being investigated under the Alternate Technology Program. Some of the treatment technologies being considered in the EIS, such as the Department's preferred melt and dilute technology for treating aluminum-based spent fuel, would require the construction of a facility to accommodate the treatment equipment, as

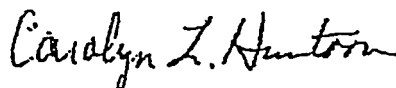
well as a dry storage facility to store the treated spent fuel pending shipment offsite.

Decisions on treatment technologies and whether to build treatment and storage facilities must necessarily await completion of the EIS. It is now apparent, however, that the design and construction of such a facility would need to be developed in the context of constrained, out-year budget targets, and funding for such a facility would need to be balanced against other priorities at SRS. We are in the process of developing a schedule that can be used as a baseline for near-term planning and budgeting purposes. The SRS Manager, Greg Rudy, has directed the SRS operating contractor, Westinghouse Savannah River Company, to identify the actions required to start melt and dilute operations in FY 2008. Once the FY 2000 funding for SRS is established, and the FY 2001 budget outlook is better understood, DOE will prepare an updated schedule for completion of the remaining technology development efforts, as well as for the design and construction of any facilities needed to implement decisions made under the EIS.

In a related development, we have received a report from the Defense Nuclear Facilities Safety Board relating to aluminum research reactor spent nuclear fuel disposition that discusses several issues previously identified by the SRS Alternate Technology Program team. These issues will be addressed during the course of the program. We do not anticipate that resolution of these issues will delay completion of the program's research and development. No additional delays in the project are contemplated as a result of the Board's report.

Please be assured that the Department remains fully committed to the investigation of alternative technologies for disposition of aluminum research reactor fuel. I will be happy to meet with you to discuss our plans. Please contact Mr. David Huizenga at 202-586-5151 to make arrangements for such a meeting.

Sincerely,



Carolyn L. Huntoon
Assistant Secretary for
Environmental Management

Comment Letter L-17

Defense Nuclear Facilities Safety Board

June 8, 1999

June 8, 1999

Mr. James M. Owendoff
Acting Assistant Secretary of
Environmental Management
Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585-0113

Dear Mr. Owendoff:

The Department of Energy (DOE) is preparing to select a processing alternative for disposal of aluminum spent nuclear fuel from research reactors, which is to take place at the Savannah River Site (SRS). DOE is proposing a melt and dilute process that does not appear to be based on a comprehensive consideration of safety, technical certainty, and cost.

The enclosed Board's staff report evaluates the spent fuel inventory at SRS and the processing alternatives for its disposal. The report is provided for further consideration of this matter as DOE proceeds in its selection of a disposition alternative. The report concludes that DOE should look for ways of accelerating the consolidation of aluminum spent fuel at SRS and maximize the use of its existing processing capabilities, at least until the year 2010, to dispose of this unique spent fuel. As more information is gained regarding the expected type and quantity of research reactor spent fuel that will be received after 2010, DOE can better evaluate alternatives for continued disposal of this spent fuel.

Although progress has been made, DOE still has a considerable amount of nuclear materials that require stabilization and either storage or disposal. SRS's processing canyons are serving as workhorses in accomplishing this important task. During the last few years, these facilities have stabilized an impressive amount of plutonium solutions, plutonium residues, plutonium production targets, and spent fuel. On the other hand, the development of new technologies for stabilization of nuclear materials has been plagued with technical problems, schedule delays, and cost overruns. Examples include the project to stabilize spent fuel at Hanford and to stabilize americium-curium solutions at SRS. Where appropriate, DOE should protect and capitalize on its existing capability where safety assurance has already been demonstrated rather than hastening to replace it. Safety of speculative technology is yet to be demonstrated.

Sincerely,

John T. Conway
Chairman

c: Mr. Mark B. Whitaker, Jr.
Mr. Greg Rudy

Enclosure

DNFSB/TECH-22

**Savannah River Site
Spent Nuclear Fuel**

Defense Nuclear Facilities Safety Board

Technical Report



April 21, 1999

Savannah River Site Spent Nuclear Fuel

J. Kent Fortenberry

EXECUTIVE SUMMARY

A large inventory of aluminum spent fuel from research reactors is being stored in water basins at the Savannah River Site. More research reactor spent fuel is to be added to this inventory. Most of this spent fuel is in better condition than the damaged aluminum defense-related spent fuel currently being stabilized at the Savannah River Site, and arguably does not require immediate stabilization. However, some of the research reactor fuel is damaged or is in a form that merits expedited stabilization. In addition, even undamaged aluminum spent fuel is not suited for long-term wet storage, both because of the unpredictability of aluminum corrosion and because of the continuous water conditioning required to protect the fuel. Furthermore, aluminum fuel, especially the high-enrichment fuel typical of research reactors, must be packaged in a manner that will be acceptable for ultimate disposal in a repository. For these reasons, the Department of Energy (DOE) must remove this fuel from wet storage and is proposing a process to prepare it for stable interim storage and final disposal.

The Defense Nuclear Facilities Safety Board (Board) provides technical safety oversight of defense nuclear facilities. The aluminum spent fuel from research reactors is not considered defense-related, but the storage and processing of this fuel will occur in defense nuclear facilities. Potential safety impacts on facility operations and decommissioning have prompted the Board to evaluate DOE's proposed actions regarding this fuel.

DOE has considered three processing alternatives for this fuel: conventional processing, melt and dilute, and direct codisposal. Conventional processing is an existing capability that involves aqueous chemical separation to produce enriched uranium for use as commercial reactor fuel and a high-level waste stream compatible with the current high-level waste processing strategy. The proposed melt and dilute process would melt the spent fuel together with depleted uranium to produce a low-enriched spent fuel melt and a waste stream from the volatilized fission products and melt crucibles. The proposed direct codisposal process would dry and package the spent fuel to produce canisters of dried spent fuel and a waste stream resulting from the cutting and cropping of the spent fuel elements. DOE's preferred alternative for most of the fuel is the melt and dilute process, to be installed in the existing L-Reactor facility at the Savannah River Site.

Melt and dilute is a new processing technology that requires significant development work. Implementation of this process depends on successful demonstration of the technology and completion of design, construction, and startup activities. DOE has had several disappointing experiences recently with the implementation of new technology. The In-Tank Precipitation and Americium-Curium Vitrification projects at the Savannah River Site are examples of new processing technologies that appeared very promising in the laboratory and even in the demonstration phase, but are proving very difficult to implement.

There are potential safety risks associated with operating the melt and dilute process in the existing L-Reactor facility. The process will involve molten spent nuclear fuel and volatilized fission products without the benefit of a canyon structure for confinement. Also, fuel-melting operations in the L-Reactor facility could result in significant facility contamination that would impact equipment operation and maintenance and introduce new challenges for decommissioning of the facility.

DOE's cost comparisons of processing alternatives are based on life-cycle costs from 1998 through 2035. For the conventional processing alternative, this life-cycle cost includes a new facility to process residual fuel receipts after 2010, when the existing canyon capability is assumed no longer to be viable. Even with this new facility added to the cost of conventional processing, DOE concludes there is no significant difference in life-cycle costs among the alternatives.

All of the existing aluminum spent fuel from research reactors will be available for processing prior to 2010. As more information is gained regarding the type and quantity of fuel that might be generated after the year 2010, DOE will be better able to evaluate future processing needs. From this perspective, conventional processing would provide additional flexibility to accommodate the potential changes in the size, type, or quantity of fuel that might be received after 2010.

After examining the aluminum spent fuel inventory and the processing alternatives, this report concludes that DOE ought to accelerate consolidation of aluminum spent fuel at the Savannah River Site and take advantage of its operating conventional processing facilities to place this fuel in stable interim storage and prepare it for final disposal. This course of action would eliminate the uncertainty of a major technology development, design, and construction effort and avoid new safety risks associated with operation and decommissioning activities in the L-Reactor facility. This course of action also would allow future flexibility by providing for stable storage and final disposal of these fuel inventories through 2010, while making it possible to defer decisions that are currently based on assumed fuel receipts 35 years into the future. When more information is available on the type and quantity of aluminum spent fuel from research reactors to be received between 2010 and 2035, DOE will be able to better evaluate its need for future processing of this fuel. More generally, this report concludes that DOE ought to concentrate its limited resources on utilizing its existing stabilization and processing capability, which is currently playing a vital role in the successful stabilization of nuclear materials.

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1. INTRODUCTION

A large inventory of aluminum spent nuclear fuel is stored in water basins at the Savannah River Site (SRS). This spent fuel was generated by the production of defense nuclear materials at the SRS reactors and by the return of spent fuel from both domestic and foreign research reactors. The generation of aluminum defense-related spent fuel at SRS has ended, but the return of aluminum spent fuel from domestic and foreign research reactors will continue. Historically, most of the aluminum spent fuel generated at SRS or received from off site was processed to recover the enriched uranium. The aluminum spent fuel received from off site was stored in the Receiving Basin for Offsite Fuel (RBOF) before being sent to canyon processing facilities. With the cessation of routine canyon processing of aluminum spent fuel from research reactors and with continued returns from off site, the aluminum spent fuel inventory has now exceeded the RBOF capacity, necessitating storage of the fuel in the L-Reactor defense nuclear facility storage basin.

The spent nuclear fuel at SRS can be categorized as either aluminum defense-related spent fuel, nonaluminum spent fuel, or aluminum spent fuel from research reactors. To date, DOE has made decisions regarding the stabilization and disposal of the aluminum defense-related spent fuel and the nonaluminum spent fuel: the defense-related spent fuel is currently being processed in the SRS H-Canyon facility, while the nonaluminum spent fuel is to be shipped to the Idaho National Engineering and Environmental Laboratory (INEEL) and consolidated with similar nonaluminum spent fuel at that site. DOE is now preparing to make a decision regarding the stabilization and disposal of the aluminum spent fuel from research reactors.

The aluminum spent fuel from research reactors differs from spent fuel from commercial reactors. The aluminum fuel is less robust and less resistant to corrosion; some is in powder form. Also, the uranium-235 enrichment of the research reactor fuel is considerably higher than that of commercial fuel. These differences pose unique challenges to the stabilization and disposal of this fuel.

DOE has prepared an SRS Spent Nuclear Fuel Management Environmental Impact Statement to assist in the selection of a processing alternative for achieving stable interim storage and disposal of aluminum spent fuel from research reactors. A Record of Decision is expected to be issued by April or May 1999. Of the eight processing alternatives considered in this environmental impact statement, three have merited extensive evaluation: melt and dilute, direct codisposal, and conventional processing. The purpose of this report is to examine the inventory of aluminum spent fuel from research reactors and the alternatives being considered for its processing.

Section 2 of this report reviews the spent fuel inventory at SRS, while Section 3 summarizes its storage. Section 4 provides a brief history of DOE's decision making on the stabilization, storage, and disposal of SRS spent fuel. Section 5 describes the melt and

dilute, direct codisposal, and conventional processing alternatives. Section 6 provides a comparison of these processing alternatives, along with a cost comparison. Finally, Section 7 presents a summary and conclusions.

2. SPENT FUEL INVENTORY

As noted in Section 1, the SRS spent nuclear fuel inventory can be divided into three categories: aluminum defense-related spent fuel, nonaluminum spent fuel, and aluminum spent fuel from research reactors. The current inventory of these categories of spent fuel at SRS is shown in Table 1. Table 1 also presents the projected additions to this inventory from off site receipts through 2035.

2.1 Aluminum Defense-Related Spent Fuel

The aluminum defense-related spent fuel, consisting of Mark 16 and 22 spent fuel assemblies, is composed of a highly enriched uranium-aluminum alloy with aluminum cladding. Long-term wet storage of this spent fuel, aggravated by poor water chemistry control and galvanic coupling, has resulted in significant pitting corrosion damage. The Defense Nuclear Facilities Safety Board (Board) addressed the need to expedite stabilization of this fuel in Recommendation 94-1 (Defense Nuclear Facilities Safety

Table 1. Savannah River Site Spent Fuel (Current Inventory and Projected Additions)

Category	Current Inventory		Additions Through 2035	
	No. of Items*	Weight (MTHM)**	No. of Items*	Weight (MTHM)**
Aluminum Defense-Related Spent Fuel	1,800	7	0	0
Nonaluminum Spent Fuel	1,200	20	0	0
Aluminum Research Reactor Spent Fuel	5,600	20	23,000	24
Total	8,600	47	23,000	24

* "Item" generally refers to a fuel element. The unique configurations of the various types of defense-related and research reactor spent fuel prevent use of a consistent unit.

** MTHM = metric tons of heavy metal (meaning the amount of uranium, thorium, or plutonium).

Source: B. Clark and C. Brizes, DOE-Savannah River Operations (personal communication, June 29, 1998); Westinghouse Savannah River Company (January 1998).

Board, May 26, 1994). The Board advised DOE that the best alternative for accomplishing this stabilization was conventional processing using the SRS canyon facilities (Defense Nuclear Facilities Safety Board, November 1, 1995). Although dry storage followed by direct disposal in a repository was evaluated for this fuel, the time required to develop new processes and facilities, as well as the technical uncertainty of the acceptability of the resulting dry fuel for ultimate disposal, led DOE to conclude that the best course of action was to process the fuel in the SRS canyons (Defense Nuclear Facilities Safety Board, August 3, 1995; U.S. Department of Energy, October 20, 1995, and February 21, 1996).

Processing of the 1800 assemblies of aluminum defense-related spent fuel (i.e., the Mark 16 and 22 fuel assemblies) began in July 1997 and is scheduled to be completed in early 2002. The stabilization process chosen for this fuel was well defined and utilized proven processes in existing facilities. In addition, ultimate disposal is ensured by utilizing currently operating processes and facilities to produce a vitrified glass waste form that is qualified for disposal in a geologic repository. The progress made thus far in stabilizing the defense-related spent fuel can be attributed to the fact that significant technology development and subsequent facility design and construction were not required. The enriched uranium recovered from the processing of this fuel will be isotopically diluted to about 5 percent enrichment and used as feed for commercial fuel.

2.2 Nonaluminum Spent Fuel

The nonaluminum spent fuel consists of a large variety of stainless-steel and zirconium-clad fuel. The materials of construction are important because they are much more durable than aluminum and more resistant to corrosion. These materials are also resistant to the chemical processes currently used at the SRS canyons. In the past, stainless-steel spent fuel from research reactors was processed at SRS using an electrolytic dissolver in the H-Canyon. This equipment is still installed, but would require some upgrades to allow resumption of operations. Zirconium-clad fuel was never processed at SRS, but mechanical operations could be employed to expose dissolvable fuel meat to the dissolution chemicals used in the canyons. Without facility modification to provide either of these mechanical operations or an alternative dissolution process, these fuels cannot be processed through the SRS facilities.

DOE has decided to consolidate similar fuel types for interim storage and preparation for ultimate disposal (U.S. Department of Energy, April 1995 and May 30, 1995). Nonaluminum fuel is to be consolidated at INEEL. All nonaluminum spent fuel stored at SRS is to be shipped to INEEL. Similarly, aluminum spent fuel is to be consolidated at SRS, and all aluminum spent fuel currently stored at INEEL is to be shipped to SRS. Interim safe storage and ultimate disposal of nonaluminum spent fuel are

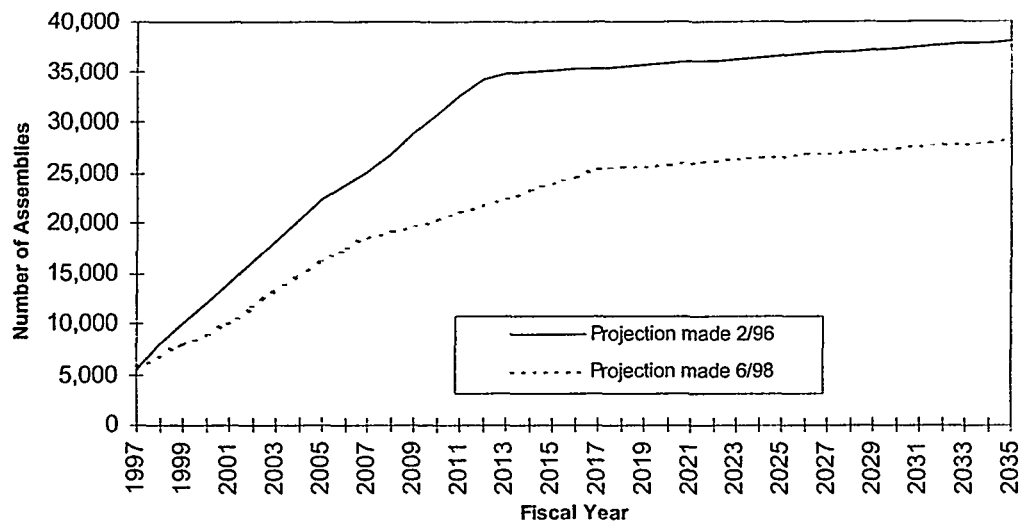
being pursued at INEEL. The alternatives being evaluated and their potential for success are not considered in this report.

2.3 Aluminum Spent Fuel from Research Reactors

The aluminum research reactor spent fuel comes from both foreign and domestic research and test reactors. A great variety of physical forms, sizes, and enrichments is associated with this spent fuel. Most of this fuel is of the Materials Test Reactor type and consists of uranium-aluminum, uranium oxide-aluminum, or uranium silicide-aluminum alloy with aluminum cladding. There is also some fuel that is of similar construction to the Materials Test Reactor type, but in differing shapes and sizes. A significant amount of fuel is in the form of loose uranium-oxide powder stored underwater in aluminum cans. There is also uranium and thorium metal fuel, either clad in aluminum or declad and packaged in aluminum canisters. A large portion of the aluminum spent fuel from research reactors is highly enriched, with a uranium-235 enrichment ranging from 20 percent to more than 90 percent.

The total SRS inventory of aluminum spent fuel from research reactors consists of the existing inventory at SRS, the projected inventory to be received from foreign and domestic research reactors, and the inventory of aluminum spent fuel to be received from INEEL. Figure 1 shows the projected total inventory through 2035. Examination of this figure reveals two significant points.

First, current projections of the total SRS inventory of this fuel are significantly lower than earlier estimates. The expected receipts from both foreign and domestic research reactors have been reduced. Foreign receipts have been reduced by about 40 percent because several foreign operators of research reactors (Belgium, France, Iran, Pakistan, South Africa, and Netherlands-HFR Petten Reactor) have opted not to return their spent fuel to DOE. Two other foreign reactor operators (Canada and Netherlands-Delft Reactor) have expressed uncertainty and are evaluating whether to return their fuel. Should these two additional reactor operators not participate, the amount of foreign research reactor fuel expected to be received at SRS would be reduced by almost 70 percent from that originally projected. Moreover, a significant number of the remaining foreign reactor operators expected to return spent fuel to the United States have requested to delay their participation until the end of the DOE program (i.e., to not return fuel until 2006 to 2009). It would not be improbable between now and 2009 for some of these remaining countries to elect not to return their research reactor fuel to DOE.



Source: B. Clark and C. Brizes, DOE-Savannah River Operations (personal communication, June 29, 1998); Westinghouse Savannah River Company (February 1996).

Figure 1. Inventory of Aluminum Spent Fuel from Research Reactors, Savannah River Site

The amount of domestic research reactor fuel expected to be received at SRS has also been reduced by about 40 percent from earlier estimates. Unlike the foreign research reactors, the domestic reactors have no alternative to returning their spent fuel to DOE. The 40 percent reduction in the amount of the domestic fuel is due instead to the permanent shutdown of several reactors. Given the age and use of many of the domestic research reactors, there are likely to be additional reductions in the fuel expected to be received at SRS from domestic sources.

The second point illustrated by Figure 1 is that most of the projected SRS inventory of aluminum spent fuel from research reactors is to be received during the first half of the planning period. Figure 2 illustrates the currently projected yearly receipts of this fuel at SRS. For the fuel from domestic reactors, the backlog is expected to be received at SRS by the year 2000. The remaining projected receipts from these reactors through 2035 are based on long-range projections from only a handful of reactors. All of the foreign inventory is to be received by 2009. Finally, the aluminum spent fuel stored at INEEL is currently scheduled to be received at SRS during 2010–2017.

Scheduling of the shipment of aluminum spent fuel from INEEL to occur after the foreign shipments have been completed in 2009 represents an attempt to level out the receipt of spent fuel at SRS. A reduction in the projected returns from foreign research

reactors, as discussed above, or a more aggressive rate of receipt would allow the INEEL shipments to be made earlier. This in turn would cause the fuel receipt profile to be even more exaggerated toward the early years of the planning period. Figure 3 illustrates the result of accelerating the shipment of spent fuel from INEEL to SRS to occur during 2002–2009.

Both Figures 2 and 3 show a small but protracted ‘tail’ of fuel receipts projected out to the year 2035. This tail is based on the estimated future generation of spent fuel from the following domestic research reactors:

- University of Missouri Research Reactor
- University of Michigan Ford Nuclear Reactor
- Massachusetts Institute of Technology Research Reactor
- National Bureau of Standards Reactor
- Brookhaven Medical Research Reactor
- Brookhaven High Flux Beam Reactor
- Oak Ridge High Flux Isotope Reactor

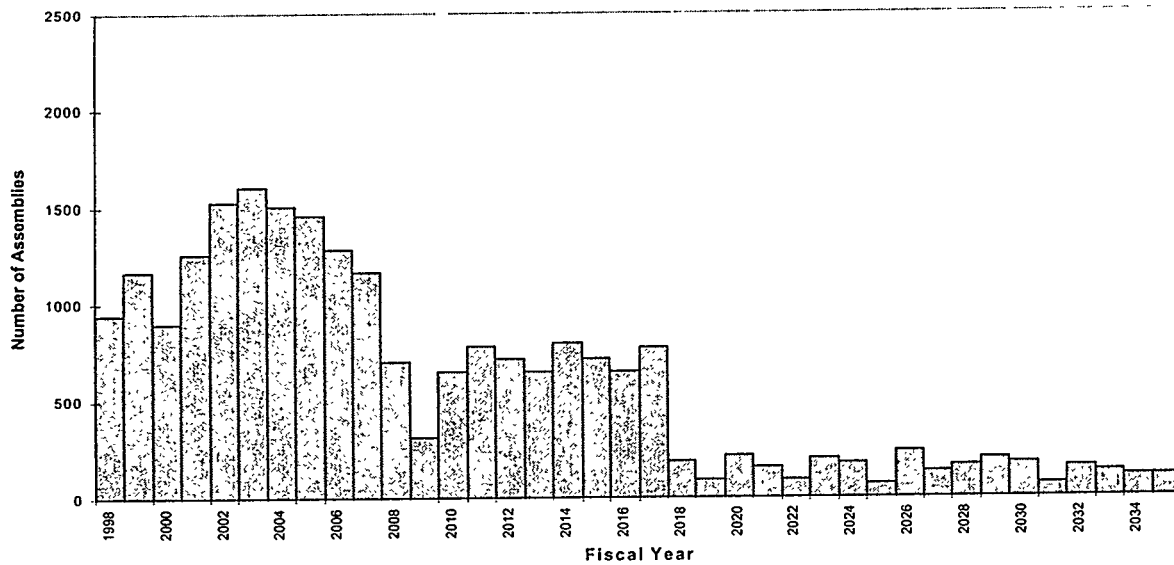


Figure 2. Receipt of Aluminum Spent Fuel at SRS

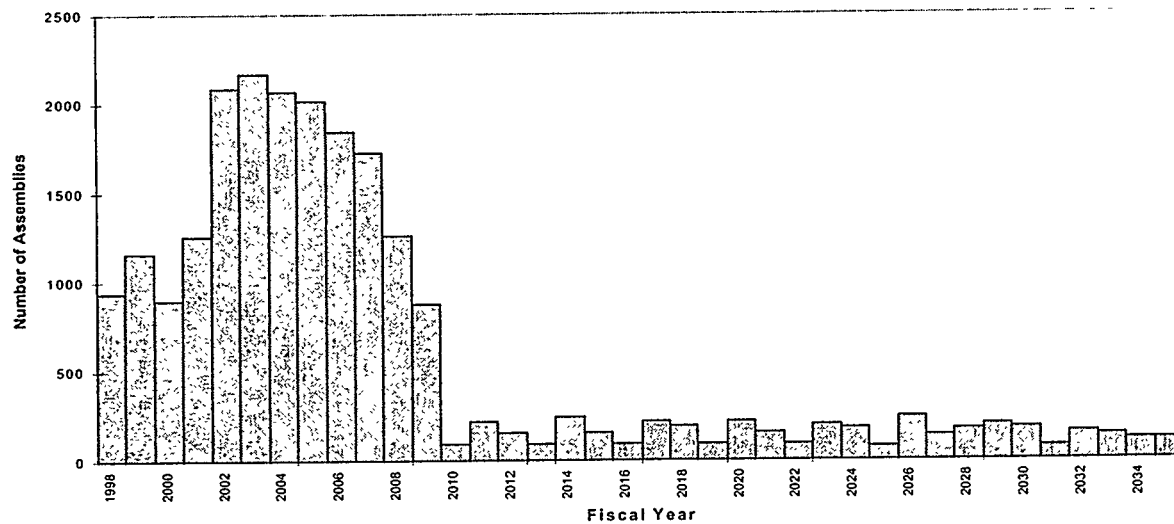


Figure 3. Receipt of Aluminum Spent Fuel (accelerated receipts from INEEL)

The above inventory projections assume constant production of spent fuel from these seven domestic research reactors through 2035. There is considerable uncertainty in this assumption. The age of these reactors currently ranges from about 30 to 40 years; by 2035, the age of these reactors will range from about 70 to 80 years. In addition, almost half of the spent fuel projection making up the tail in Figures 2 and 3 is from the Brookhaven High Flux Beam Reactor, which is currently shut down. The decision on whether to restart this reactor is pending.

The above projections of future fuel receipts affected the way DOE approached the evaluation of alternatives for its aluminum spent fuel. DOE assumed that the existing canyon processing capabilities at SRS would not be maintained beyond the year 2010. With this constraint, and with significant fuel receipts projected for 2010–2035, the conventional processing alternative would have to utilize canyon processing until 2010 and then implement a new replacement treatment capability. It is interesting to note that even under this scenario, the conventional processing alternative remains financially competitive with the melt and dilute and direct codisposal alternatives.

As discussed earlier, the amount of spent fuel projected toward the end of the planning period may be grossly overestimated. Also, 10–15 years from now, the fuel type used in a refurbished 50-year-old research reactor or in a new replacement research reactor may not be compatible with any of the processing alternatives currently being considered. DOE's decision process appears to have been too heavily influenced by these questionable projections of fuel inventories 15–40 years from now.

3. SPENT FUEL STORAGE

Interim management of spent nuclear fuel at SRS consists of storage in three water-filled storage basins: K-Reactor storage basin, L-Reactor storage basin, and RBOF.

The inventory in the K-Reactor storage basin consists of Mark 16 and 22 aluminum defense-related spent fuel. The K-Reactor storage basin is expected to be deinventoried of spent fuel by the end of 2000, when the last of its inventory is transferred to the H-Canyon for processing.

The inventory in the L-Reactor storage basin consists of Mark 16 defense-related spent fuel plus the research reactor spent fuel that has been placed in storage there since 1997. By the end of 2001, all of the Mark 16 spent fuel will have been transferred to the H-Canyon for processing. The L-Reactor storage basin will then continue to store aluminum spent fuel received from both foreign and domestic research reactors.

The RBOF inventory consists of a large variety of both aluminum and nonaluminum research reactor spent fuel. The nonaluminum fuel is scheduled to be transferred to INEEL. RBOF will then continue to store the aluminum fuel.

Several vulnerabilities have been identified with wet storage of spent fuel at SRS. Severe water chemistry problems and significant fuel corrosion have been experienced periodically in the K- and L-Reactor storage basins. The Board addressed the susceptibility of aluminum spent fuel to corrosion in DNFSB/TECH-7, *Stabilization of Deteriorating Mark 16 and Mark 22 Aluminum-Alloy Spent Nuclear Fuel at the Savannah River Site*, which led to expedited stabilization of corroded defense-related spent fuel in the H-Canyon. In addition, to improve storage conditions pending stabilization, the water quality and chemistry control capabilities at both the K- and L-Reactor storage basins were recently upgraded.

The RBOF facility has historically maintained very good water chemistry. Even so, some especially susceptible aluminum spent fuels stored at RBOF have experienced failure, prompting decisions to stabilize these fuels by processing them in the SRS canyons.

Successful long-term storage of aluminum spent nuclear fuel depends on maintaining high-purity water. However, corrosion behavior is difficult to predict. For example, some amount of corrosion at preexisting sites of pitting or other clad damage can continue even under excellent water chemistry conditions. Also, recent studies at RBOF have noted the presence of microorganisms and biofilms that could contribute to

microbial-influenced corrosion of the aluminum spent fuel (Westinghouse Savannah River Company, June 1, 1998).

Most of the aluminum research reactor fuel being stored at SRS is in generally better condition than the defense-related fuel stored in the reactor disassembly basins. Water and fuel conditions in RBOF and the L-Reactor storage basin are being closely monitored for signs of water chemistry changes, fuel corrosion, or microbial activity. Even so, there is no fundamental disagreement on the vulnerability of aluminum fuel in wet basin storage.

Some of the aluminum spent fuel from research reactors has already been stored in water basins for up to 40 years. A small portion of this fuel has experienced significant cladding corrosion. Some the fuel being returned from foreign reactors is sufficiently corroded to require aluminum canister overpacks before being stored in the SRS water basins. Also, a portion of the research reactor spent fuel is in a form that is particularly susceptible to corrosion during wet storage. Examples of such forms include uranium metal fuels that have been declad, fuel that has been cut up for research purposes, and loose uranium oxide spent fuels, all of which have been packaged in aluminum canisters for storage in the SRS water basins.

DOE has stated its general commitment to avoiding long-term storage of spent nuclear fuel in wet storage basins (U.S. Department of Energy, February 1996 and December 1998a). When specific cases are considered, such as damaged or declad fuel, the need to remove the fuel from long-term wet storage is even more pressing.

4. HISTORY OF DOE'S DECISION MAKING ON SPENT FUEL

The history of DOE's decision making related to the stabilization, storage, and disposal of SRS spent nuclear fuel starts with the 1992 decision to stop reprocessing of spent fuel for the purpose of producing defense nuclear materials (U.S. Department of Energy, April 28, 1992). In announcing this decision, DOE stated that phaseout plans should "include processing of existing inventories of aluminum clad fuel at SRS as well as fuel receipts while stabilization is being conducted." DOE noted that processing of the aluminum spent fuel "will result in less than a two percent increase in high-level waste" and "will also relieve potential near-term fuel storage problems...." However, because canyon processing had been shut down, the existing inventory of aluminum fuel at SRS could not be processed. Instead, the spent fuel inventory remained in underwater storage. The DOE spent fuel and related facilities were designed for reprocessing, not for long-term storage underwater, and the shift in activities from reprocessing to longer-term storage soon began to evidence problems.

Starting in March 1993, the Board's staff began reviewing the storage of DOE-owned spent nuclear fuel. In November 1993, DOE issued the *Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear Fuel and Other Reactor Irradiated Nuclear Materials and Their Environmental, Safety and Health Vulnerabilities*. This report identified existing and developing problems associated with the long-term storage of spent fuel. DOE has expended considerable resources in addressing these vulnerabilities.

The Board's Recommendation 94-1, *Improved Schedule for Remediation in Defense Nuclear Facilities Complex*, was issued in May 1994. This recommendation, focused on DOE's defense-related nuclear materials, expressed special interest in the large amounts of deteriorating spent fuel stored in canyons and reactor storage basins, and recommended that DOE expedite the processing of this fuel into a form suitable for safe storage. DOE's Implementation Plan for this recommendation was issued in February 1995 and proposed the use of conventional processing to stabilize the aluminum defense-related spent fuel at SRS.

In May 1995, DOE decided to consolidate similar fuel types. This decision resulted in plans to send all aluminum spent fuel to SRS and nonaluminum spent fuel to INEEL. One exception to this consolidation plan is the small amount of aluminum fuel from production reactors at the Hanford K-Basin facility. This fuel was thought to be best managed together with the N-Reactor fuel at Hanford. A more recent evaluation has led to the recommendation that this aluminum fuel be considered for processing in the SRS canyons (U.S. Department of Energy, May 6, 1998).

In December 1995, February 1996, and April 1997, DOE issued decisions to use canyon processing to stabilize a significant amount of SRS aluminum fuel. These decisions resulted in successful processing of about 16,000 aluminum defense-related Mark 31 targets, successful processing of all of the aluminum spent fuel from the Taiwan Research Reactor stored at SRS, and successful processing of a leaking canister of Experimental Breeder Reactor-II spent fuel. In addition, these decisions led to the current processing of the aluminum defense-related Mark 16 and Mark 22 spent fuel.

In May 1996, DOE issued a Record of Decision (ROD) for the Foreign Research Reactor Environmental Impact Statement (FRR EIS), establishing a 10-year policy to accept and manage aluminum spent fuel from foreign research reactors. The return of this spent fuel would result in a substantial amount of spent fuel at SRS. To avoid continued long-term wet storage of this fuel, the FRR EIS attempted to select a process for establishing a stable long-term storage configuration and ultimately a path for the fuel's disposal. Conventional processing was an alternative for accomplishing this stabilization and disposal, but one premise of the FRR EIS was avoidance of conventional processing as a result of nonproliferation concerns.

Because ready alternatives to conventional processing were not available, the ROD proposed that the development of alternative technologies be initiated. The undesirability of continued wet basin storage of the aluminum spent fuel while these alternative technologies were being developed prompted DOE to limit the development time; if the new technology could not be implemented by 2000, conventional processing would be used. In addition, because some of the aluminum fuel was known to be damaged and especially vulnerable to wet basin storage, the ROD provided for conventional processing of any aluminum spent fuel from foreign research reactors that presented a health and safety concern. To help in the decision making on how to stabilize and dispose of the aluminum spent fuel, the ROD proposed the conduct of additional studies of the proliferation risks, costs, and timing associated with conventional processing.

DOE has prepared an SRS Spent Nuclear Fuel Management Environmental Impact Statement (SNF EIS) to assist in selection of a processing alternative for achieving stable interim storage and disposal of the aluminum spent fuel. A ROD is expected to be issued by April or May 1999. The SNF EIS represents an attempt to complete the decision-making process for determining the stabilization and disposal of the remaining aluminum spent fuel at SRS. As noted earlier, although DOE presents eight alternatives in the SNF EIS, only three of these have merited close examination: melt and dilute, direct codisposal, and conventional processing.

DOE has sponsored several studies and reviews whose results will be factored into its decision on a processing alternative. Some of these are discussed below.

Research Reactor Spent Nuclear Fuel Task Team Report (U.S. Department of Energy, May 1996 and June 1996). As proposed by the FRR EIS ROD, DOE sponsored a task team to evaluate eleven potential alternative technologies for processing of aluminum spent fuel. In the final draft of this report, conventional processing was used as a baseline and was scored significantly higher than the alternative technologies in terms of confidence in success, cost, technical suitability, and timeliness. However, in the final version of this report, conventional processing was removed, and only the relative scores of the alternative technologies were given. The report concluded that direct codisposal should be pursued as the primary processing alternative, with melt and dilute (or press and dilute) used as a backup.

The task team report also identified certain aluminum domestic spent fuels the team believed should be processed using the conventional processing capability at SRS, regardless of the alternative technology selected. These included metal fuel (especially susceptible to corrosion), particulate or powdered fuel (readily dispersible, requiring special handling, and difficult to qualify for direct disposal), and one-of-a-kind fuel (requiring special handling or canning). These fuels were listed in Table 5.2-1 of the final report, and so have become known as the Table 5.2-1 fuel. DOE is proposing to follow this recommendation and process the Table 5.2-1 fuel, which represents about 3 percent (by volume) of the total projected SRS inventory of aluminum research reactor fuel, in the SRS canyons. A similar category of fuel, identified as Table 5.2-2 fuel, consists of spent fuel from foreign research reactors expected to be received in powdered form. The Table 5.2-2 fuel, which represents a significant amount of powdered fuel, is still planned to undergo DOE's proposed melt and dilute process. The distinction between the Table 5.2-1 powdered spent fuel and the Table 5.2-2 powdered spent fuel appears to be arbitrary.

Spent Nuclear Fuel Alternative Technology Decision Analysis (Westinghouse Savannah River Company, June 26, 1998a). In this analysis, Westinghouse Savannah River Company (WSRC) weighed the relative merits of the direct codisposal and melt and dilute alternatives and recommended that DOE use the melt and dilute process to replace conventional processing. However, WSRC also suggested that uncertainty in repository requirements made it prudent to continue development of the direct codisposal alternative as a backup technology. The conventional processing alternative was not considered in this analysis.

National Research Council Review (National Research Council, 1998). This review evaluated the alternatives being considered to replace conventional processing. The report generally concluded that DOE was seeking a definitive solution to a problem that was not yet well defined, before the information needed to make sound choices was available. DOE was therefore encouraged to apply a phased strategy for selection and implementation of a processing alternative. The report pointed out the uncertainty in future fuel receipts; the poorly defined waste acceptance requirements; and the

uncertainties of cost, performance, and safety of unproven technologies. The report also concluded that there appeared to be no technical basis for rejecting conventional processing and that "DOE should have given more careful consideration to the conventional reprocessing option for treating aluminum spent fuel" (p. 7). The report stated that conventional processing "has been demonstrated for aluminum spent fuel ..., the costs and risks are well known, the necessary facilities are currently in operation at Savannah River, and the waste form (borosilicate glass) will likely be acceptable for disposal at the repository" (p. 7).

DOE-Sponsored Multi-Attribute Utility Decision Analysis (Sandia National Laboratories, May 23, 1998). This decision analysis, conducted at Sandia National Laboratories, weighed the relative merits of two of the DOE alternatives: direct codisposal and melt and dilute. The conclusion reached was that the melt and dilute alternative is preferable to direct codisposal. Traditional canyon processing was not considered in this analysis.

Nuclear Regulatory Commission Review (Nuclear Regulatory Commission, June 5, 1998). DOE requested that the Nuclear Regulatory Commission conduct a topical review of the melt and dilute and direct codisposal alternatives. The report resulting from this review concluded that although additional development work was required, both melt and dilute and direct codisposal would be acceptable concepts for the processing of aluminum research reactor spent fuel. The issues identified as requiring additional work included the degradation of the fuel during interim storage in the road-ready canister; postclosure criticality; and the partitioning of radionuclides in melted fuel among slag, alloy, and offgas. Canyon processing was not considered in this review.

Westinghouse Savannah River Company Cost Study (Westinghouse Savannah River Company, December 1997 and May 1998; and U.S. Department of Energy, December 1998b). This study compared annual, 10-year, and life-cycle costs for variations on the direct codisposal, melt and dilute, and conventional processing alternatives. As described earlier, all of the variations on conventional processing call for canyon processing until 2010, followed by use of a new treatment capability. Even so, the conventional processing variations were found to be cost-effective, especially during the first decade. One exception was an alternative involving the construction of a new conventional processing facility.

Nonproliferation Assessment (U.S. Department of Energy, December 1998a). This assessment was performed to meet the commitment for an "independent study of the nonproliferation...implications of chemical separation of spent nuclear fuel from foreign research reactors" made in DOE's ROD for the FRR EIS. Rather than limit the scope to chemical separation, this study evaluated the nonproliferation implications of all technology alternatives being considered by DOE for stabilization and disposal of the aluminum spent fuel. This assessment reached several conclusions, including that "all of

the options would be consistent with U.S. nonproliferation policy....” (p. ES-11), and “all of the options have the potential to support fully U.S. efforts to reduce the civil use of HEU....” (p. ES-11). However, this assessment generally concluded that the Office of Arms Control and Nonproliferation “views the melt and dilute recommendation as a favorable technology in light of nonproliferation concerns. A decision to reprocess a majority of the aluminum-based spent fuel at the Savannah River Site could negatively affect the credibility of U.S. policy not to encourage reprocessing. Such a decision would also extend the period of time that reprocessing operations must continue at the Savannah River Site - making it more difficult for U.S. efforts to convince other nations not to pursue fuel cycles that increase proliferation risks” (p. 4.2).

5. DESCRIPTION OF PROCESSING ALTERNATIVES

This section describes the three processing alternatives that have merited close examination: melt and dilute, direct codisposal, and conventional processing.

5.1 Melt and Dilute

The proposed melt and dilute alternative involves melting a mixture of spent fuel and depleted uranium to reduce both volume and enrichment. The current proposal for implementing this process locates the melt and dilute operation within the existing L-Reactor facility. Melt and dilute is a new technology, so the following description of the process is largely conceptual.

All of the spent fuel is assumed to be stored or staged in the L-Reactor storage basin. Spent fuel that is not already stored in the L-Reactor storage basin, such as that stored at RBOF or not yet received on site, is assumed to be transported, received, and unloaded into the L-Reactor storage basin.

Pretreatment characterization requirements are unknown at this time. Although extensive analysis may be required, the sampling and analysis of the melted fuel would be much more straightforward than characterizing the large variety of intact fuel elements.

The aluminum spent fuel is stored in various arrangements (e.g., shipping cans, bundles, tubes). This fuel is disassembled as required, and some portions of the nonfuel aluminum are cropped or removed by means of underwater cutting. Some fuel types, such as the High Flux Isotope Reactor cores, require special size reduction, accomplished by cutting the fuel area or by crushing. The cropped and sized fuel is transferred into the L-Reactor processing area using an existing transfer canal. Once in the processing area, the fuel is staged in drip-dry fuel storage racks to await the melting operation. From the staging racks, the fuel is transferred to a drying oven or rack, where an elevated air temperature (200°C) is maintained to preheat and further dry the fuel. All liquid water must be removed from the fuel to prevent a melt-water interaction or steam explosion that could expel molten fuel from the furnace. Additional puncturing and drying activities are necessary to accommodate aluminum canisters or overpacks that may be used to handle cut-up fuel pieces or loose oxide. These canisters require special care to prevent pressurization in the furnace and to ensure that all liquid water is eliminated.

The dried spent fuel is charged into a furnace crucible, melted, heated to above the alloy liquidus temperature, and stirred. A sample is drawn from the melt and transferred out of the process area for quick analysis of total uranium concentration and uranium isotopic concentration. Based on sample results, appropriate amounts of depleted

uranium and aluminum are added to the crucible to achieve both the desired uranium-235 enrichment and the desired alloy composition. This 'recipe' is based on the sampling results, as well as on the type of fuel (i.e, uranium-aluminum alloy, uranium-silicide, or uranium-oxide). Preliminary criticality evaluations indicate that the addition of neutron poisons may also be required. The added materials are melted, and the mixture is maintained at a temperature of at least 850°C, and possibly 1000°C or higher. Induction currents are used to stir the mixture. At elevated temperatures, some of the fission products volatilize. The furnace is maintained at negative pressure by an offgas system that captures the fission products (with the exception of noble gases, such as krypton) and exhausts to an existing ventilation stack.

A second sample is drawn from the melt and transferred out of the process area for composition analysis. This second sample confirms the final uranium concentration and alloy composition, as well as providing other necessary characterization data. The melted mixture is allowed to harden in the crucible.

When the fuel melt is solid and cooled, it is removed from the crucible, conveyed to a shielded transfer cell, and loaded into a shielded canister. The canister is about 17 inches in diameter and about 120 inches long and accommodates several fuel melts. Using a shielded transfer cask, the filled canister is conveyed to a canister preparation station where the canister is backfilled with helium, welded closed, and leak tested. Finally, the transfer cask is loaded onto a transporter and transferred to modular dry storage units outside the L-Reactor facility. The fuel is stored on site in these modular dry storage units pending transfer to the geologic repository.

5.2 Direct Codisposal

The proposed direct codisposal alternative involves packaging the spent fuel without changing the fissile material enrichment and without significantly altering the volume. As with the melt and dilute alternative, packaging operations are located within the existing L-Reactor facility. Also as with the melt and dilute process, the following description is largely conceptual.

All of the spent fuel is assumed to be in storage at the L-Reactor water storage basin. Spent fuel that is not already stored in the L-Reactor water storage basin, such as fuel stored at RBOF or fuel not yet received on site, is assumed to be transported, received, and unloaded into the L-Reactor water basin.

Pretreatment characterization requirements are currently unknown, but will be based on repository acceptance criteria, which are still being defined. Depending on the data determined necessary to meet repository requirements, characterization activities could be extensive. Data from fuel shipping papers do not always meet repository quality

assurance requirements. Also, in contrast with commercial fuel, the property data and operation history for research reactor fuel range from good to nonexistent. Initial assessments indicate that the characterization requirements for direct codisposal will be significant, involving visual, thermal, neutron, gamma, and weight measurements.

As with the melt and dilute alternative, the fuel is disassembled as required, and some portions of the nonfuel aluminum are cropped or removed by means of underwater cutting. The cropping of the fuel assemblies to eliminate most of the nonfuel structural components reduces the volume of spent fuel.

Drying of the cropped fuel might be accomplished by removing the fuel from water storage and performing a drying operation similar to the drip-dry and preheat done for the melt and dilute process. In this case, the dried fuel would then be loaded into a dry storage canister. Alternatively, the fuel might be loaded underwater into a dry storage canister, and then undergo a water removal and drying process while in the canister. In either case, the assemblies would be vacuum-dried to remove free water.

Because of the enrichment of the research reactor spent fuel, neutron poison must be added to the canister, probably incorporated into baskets that serve to separate the fuel elements. The loaded canister will likely be vacuum sealed or backfilled with helium prior to weld sealing. Drying requirements must address maximum residual free water within the sealed canister to limit corrosion and hydrogen gas generation. Acceptance testing is required to verify various attributes of the package, such as residual water content and leak tightness.

The dry storage canister has a diameter of about 17 inches and a height of about 120 inches. Three to four baskets of fuel are stacked within each canister. After external decontamination, the dried, sealed fuel is generally considered road-ready because no further characterization, conditioning, or other handling is necessary before shipping. However, depending on the storage configuration, the canisters may require packaging into shipping casks and may need venting to relieve buildup of hydrogen pressure prior to shipping.

The canister temperature is limited to 200°C during interim storage to avoid excess creep and the potential for hydrogen blistering of the aluminum fuel and cladding materials during drying and storage. The heat transfer properties of the helium backfill result in lower temperatures and reduced degradation rates. The helium backfill also provides an inert atmosphere in the event that hydrogen generation should exceed 4 percent by volume.

Several requirements related to retrievability are likely to be imposed. Retrievability requirements define "the acceptable degradation or change in condition of the direct-stored [aluminum spent fuel] form that is allowable during the interim dry

storage period. The requirement is based on engineering judgment to provide for ready removal of the fuel from a canister and handleability of the fuel" (Westinghouse Savannah River Company, October 1997, p. 3.4). Retrievability requirements address general corrosion or pitting corrosion (less than 0.003 inches in depth in spent fuel cladding or in exposed fuel material), plastic deformation of the fuel (less than 1 inch over a 3-foot length and deformation less than 75 percent of the clearance space between the fuel and the storage grid), and fuel rupture due to creep or severe embrittlement. There may also be a need for statistical sampling to ensure canister integrity and to verify compliance with waste acceptance criteria prior to transportation to a geologic repository.

One significant disadvantage of the direct codisposal alternative is that the scope of fuels to be processed is generally limited to the Materials Test Reactor design. The larger involute design (High Flux Isotope Reactor and Reacteur à Haut Flux) and longer pin design fuels cannot be accommodated, nor can the loose oxide fuel and metal fuels.

5.3 Conventional Processing

The conventional processing alternative described here is currently being used to stabilize aluminum defense-related spent fuel, as well as some of the foreign research reactor spent fuel. Because this alternative involves an operating facility and not a new technology, only a cursory description is provided.

Spent fuel, currently stored in RBOF or the L-Reactor water storage basin, is loaded in transportation casks and transferred to the H-Canyon via the site rail. At the H-Canyon, the spent fuel is unloaded and either placed into temporary staging storage or charged directly to the dissolver.

The fuel cladding and fuel meat are dissolved using hot nitric acid, which results in a solution of uranium, aluminum, fission products, and small amounts of transuranics (neptunium and plutonium). This solution is clarified by precipitation and centrifuge to remove certain impurities. It is then separated using a solvent extraction process into a waste stream, containing fission products and small amounts of transuranics, and a purified uranium stream.

The waste stream is neutralized and sent to the high-level waste storage tanks, adding to the current SRS inventory. The purified uranium stream (uranyl nitrate solution) is isotopically blended with depleted uranium solution to an enrichment of about 5 percent, and used as feed for commercial reactor fuel.

6. COMPARISON OF PROCESSING ALTERNATIVES

This report is concerned with risks to workers, the public, and the environment. The processing alternatives being proposed for the stabilization and disposal of the remaining aluminum spent fuel at SRS are examined here from the perspective of reducing the risks associated with operating the process, reducing the risk that the process might not be successful in providing for stable interim storage and ultimate disposal, and maximizing utilization of the limited resources available for risk reduction.

All of the processing alternatives considered involve risks to workers, the public, and the environment. The draft SNF EIS attempts to compare the risks associated with the various alternatives. However, the quantification of risks in the SNF EIS requires the comparison of numbers that are, because of the large uncertainties involved, generally below our ability to discriminate. For example, the estimated number of latent cancer fatalities in the SRS offsite population as a result of more than 35 years of fuel treatment is estimated to be 0.0034 for DOE's preferred alternative and 0.0044 for the maximum impact alternative. Likewise, the latent cancer probability for the maximally exposed member of the public is estimated to be 9.5×10^{-8} for DOE's preferred alternative and 3.3×10^{-7} for the maximum impact alternative. For site workers receiving no more than the allowed annual administrative radiation dose during the same period, the estimated latent cancer probability is reported as 2.8×10^{-3} .

These risk quantities are not very useful for discriminating among the alternatives. Furthermore, the risk estimates reported in the draft SNF EIS represent mitigated risk, and so include significant assumptions that may be inappropriate. For instance, the proposed melt and dilute processing alternative is to be located in the L-Reactor facility, but in estimating the release of radionuclides, the draft SNF EIS assumes that the process is implemented in a converted process cell of one of the SRS canyons (Westinghouse Savannah River Company, March 25, 1997). This assumption could result in a considerable underestimation of releases if the confinement afforded by the canyon structure and ventilation system is not achieved in the L-Reactor facility.

To compare the risk involved in operating the processing alternatives, this report emphasizes the new or unique risks that would be introduced. Although this approach results in a bias for the existing technology, it is a justified bias. Given a process with known and familiar hazards, it is possible to take advantage of previously developed experience and expertise to eliminate or mitigate these hazards. Unless a new technology offers significant risk reduction, risks that have been addressed and successfully mitigated during the course of several decades are more palatable than new risks that may not be well defined or understood and for which mitigation approaches have not yet been developed.

Besides the risks associated with operating the process, there is also risk associated with not implementing the process successfully, with experiencing substantial delay in the implementation of the process, or with failing to achieve ultimate disposal of the spent fuel. The longer the aluminum spent fuel is allowed to remain in interim storage in wet basins, the greater is the possibility of chemistry/corrosion problems, handling difficulties, criticality incidents, contaminations, and so on. Additionally, there is the risk of producing a material form that turns out to require further treatment, handling, or processing in order to meet repository acceptance requirements. Therefore, in comparing processing alternatives, this report also examines the risks and uncertainties associated with their successful implementation.

DOE faces a substantial task in stabilizing and cleaning up legacy nuclear material. The Board has focused on the stabilization of certain high-priority materials through various recommendations, especially Recommendation 94-1. However, DOE does not have unlimited resources. In general, large expenditures in one area challenge the resources available for other areas. It is prudent to compare the costs of the processing alternatives, since these costs represent resources that would be unavailable for addressing other conditions within the DOE complex. Such a cost comparison is provided at the end of this section.

6.1 Melt and Dilute

6.1.1 Risks Associated with Operating the Process

Melter Offgas. The melt and dilute process would require an offgas treatment system to remove volatile radioisotopes in the gases and vapors that would result from melting the fuel. In general, hot volatilized fission products would be condensed onto cooler surfaces. Cesium, for example, boils at 671 °C and will condense on surfaces kept below this temperature. Baffles or other media maintained at below about 600 °C could be positioned in the offgas stream to condense volatilized fission products. The condensing medium would also serve to cool the offgas in order to protect downstream high-efficiency particulate air (HEPA) filters. Radioactive particulates not collected onto the condensing medium would be captured downstream on these HEPA filters. Tritium, iodine, krypton, and other gaseous radionuclides would probably not be captured and would be released. Since the melting point for the uranium-aluminum alloy fuel is low (approximately 630 °C), the mobility of fission products in the fuel above this temperature is quite high. Very little time is required for fission products to be released from the molten fuel once the boiling point of the fission products has been exceeded. Also, higher boiling point elements can sometimes associate with more volatile species and be released at temperatures lower than their boiling point. Studies have concluded that the ideal melting point for the diluted uranium-aluminum mixture would be near 850 °C (Westinghouse Savannah River Company, October 1997). However, stirring requirements

(i.e., the need for a lower viscosity), uranium sampling methods, and dilution techniques could drive this temperature up to 1000°C or higher. At higher temperatures, the volatilization and release of fission products would increase. Release rates would also increase with stirring. In addition to the confinement of volatilized fission products during normal operation, the offgas system could be required to provide treatment and confinement during or following abnormal conditions, such as loss of offgas cooling, melter overheating, energetic events inside the melter, facility fires, loss of power, and natural phenomena hazards (e.g., earthquake, high winds).

Building Ventilation and Confinement. The proposed melt and dilute furnace incorporates an offgas treatment and confinement system that would probably be tied into the existing process area ventilation system exhaust, discharging directly into the existing L-Reactor facility ventilation stack. Given the fuel handling, drying, and furnace operations in the L-Reactor facility process area, the existing ventilation that serves this area would need to provide a confinement function. This confinement function would become especially important during process upsets. It is not clear that the L-Reactor process area ventilation system could provide the reliable confinement function needed for melt and dilute operations. At a minimum, upgrades or modifications would be required.

Steam Explosions. The introduction of water in or around the melter would result in the potential for an energetic steam explosion event. Such an event could result in pressurization or mechanical damage to the furnace or auxiliary systems, such as the offgas system, the furnace room, or the process room, leading to expulsion of material from the melter and an unfiltered release of radioactive material. Water cooling of the furnace induction coils is one source of water that is part of the melter design. Another opportunity to introduce water into the melter is during the charging of spent fuel that has not been properly dried, damaged fuel, or fuel canisters or packages that might retain water from the storage pool.

Criticality. Storage of highly enriched spent fuel poses the potential for criticality. While the fuel is in wet storage, the consequences of an inadvertent criticality are substantially mitigated by the shielding and confinement provided by the water. However, the melt and dilute process involves handling the fuel out of the basin water. Fortunately, with the melt and dilute process located in the process area of the L-Reactor facility, substantial shielding would already be in place. In addition to the handling and storage of spent fuel, the amount of highly enriched fuel being added to the melter would need to be controlled to avoid a melter criticality.

Contamination. Drip-dry lag storage racks would be used in the process area before fuel was transferred to the furnace module. The air drying of fuel in this rack could contribute to airborne contamination levels in the process area. In addition, some fraction of the fuel to be processed would be damaged. Damaged fuel would further exacerbate the release of fission products and/or fissile material during the handling of dry fuel.

Another potential source of contamination is the melter operation. A negative pressure offgas system would be utilized to confine fission products volatilized in the melter. However, spent fuel would have to be charged to the melter, samples taken, dilution material added, and the resulting fuel melt removed. A significant amount of contamination could result from the flow of materials into and out of the melter during the course of several decades. Confinement of this contamination is discussed above as building ventilation and confinement. Although the proposed melt and dilute operation would be remotely operated, the melt and dilute process is not being designed as a remote canyon type of operation. Personnel access would be required to conduct equipment maintenance and repair. The reliability and maintainability of these remotely operated melt and dilute activities are unknown. Continually increasing contamination, poor equipment performance, and the need for personnel access to maintain the equipment could present significant operational challenges.

Waste Handling. The condensing medium in the melter offgas system would need to be removed periodically for disposal or washed with acid solution to remove the collected radionuclides. The acid solution would be neutralized and transported to the tank farms as high-level waste to await vitrification at the Defense Waste Processing Facility (DWPF). "Handling and disposing of substantial quantities of material of this type is a situation that would be...a first-of-a-kind process at SRS" (Westinghouse Safety Management Solutions, Inc., February 26, 1998, p. 4). In addition, downstream HEPA filters would collect material that did not condense onto the offgas system condensing medium. Given the nature of the offgas stream, and considering process upsets such as loss of offgas cooling, these HEPA filters could also pose a challenge for handling and disposal. Wastes from the melter offgas system have been estimated at one set of used HEPA filters every 3 months and 50 gallons of high-level waste solution per month.

Furnace Material Carryover. "A small fraction of the uranium and plutonium content of each melt batch may escape from the liquid surface and eventually adsorb on the filters (or elsewhere in the offgas system or ventilation system) as metallic and oxide dust particles or fines" (Westinghouse Safety Management Solutions, Inc., February 26, 1998, p. 4). This possibility must be addressed relative to criticality and material release. An aluminum-uranium casting furnace that was operated for years at SRS for the fabrication of new fuel deposited kilogram quantities of uranium-235 within the ventilation ductwork. Also, aluminum dusts can be combustible and can pose a detonation hazard under certain conditions. There is extensive experience at SRS with aluminum-uranium foundry operations. However, there is very limited experience with foundry operations involving highly radioactive fission products.

Unique Spent Fuel. Powdered spent fuel is included in the inventory of spent fuel that would be treated with the melt and dilute process. This material represents additional unique challenges. Also included in the inventory are various configurations of canisters and storage tubes, some of which are sealed. The powdered material would need to be

repackaged, and would have to be handled and recanned to allow remote handling for charging to the melting furnace. Packages used for underwater storage would probably need to be punctured or opened to ensure that no water was introduced into the furnace and that sealed cans did not pressurize and rupture when heated in the melting furnace.

6.1.2 Risk of Not Being Successfully Implemented

Design, Construction, and Startup. Stabilization and disposal of the aluminum spent fuel depends on the ability to design, construct, and start up the facilities needed for the melt and dilute process within reasonable cost and schedule constraints. A challenge for the melt and dilute process is the need for significant technology development. Current plans are to locate the process in the L-Reactor facility. This facility was not designed for the kinds of activities required by the melt and dilute process. The shortcomings of the facility would have to be addressed in the design of a more elaborate, self-contained melter. In addition to providing its own confinement, the melter would require remote operation. The melter would have to be capable of providing at least two remotely obtained samples per charge. It would also require inductive stirring to ensure homogeneity of the fuel melt. If the melted fuel were not poured into a canister, the melter would have to remotely discharge its solidified fuel melt for subsequent packaging while maintaining confinement.

A suitable crucible furnace insert to contain the melted fuel would have to be identified. Metal crucibles tend to fail by interaction with the fuel melt at high temperatures, and graphite crucibles appear to have an affinity for some of the fission products. The ability to replace crucibles frequently would also be required.

To achieve the desired alloy composition and enrichment, each melt would have to be sampled and specific amounts of uranium and aluminum added. This process would require substantial engineering since the necessary additions would depend not only on enrichment and the amount of structural aluminum, but also on whether the fuel was a uranium-aluminum alloy, a uranium-silicide alloy, or a uranium-oxide alloy.

There is some concern that fuel pieces would have to be no longer than two-thirds the crucible diameter to prevent "bridging." If this were necessary, pretreatment for the melt and dilute process would involve substantial fuel handling and cutting operations that would in turn result in additional contamination and criticality concerns.

The above examples of technical challenges are not necessarily prohibitive. However, recent difficulties in the DOE complex with the development of unique processes for stabilizing highly radioactive materials—such as the vitrification of americium-curium at SRS, the removal of cesium from high-level waste in the In-Tank Precipitation Facility, and the stabilization of N-Reactor fuel at Hanford—illustrate the likely cost and schedule uncertainties of the melt and dilute process.

Repository Acceptance Requirements. Even if stable interim storage is achieved, the ability to produce a waste form that is acceptable for the repository is essential to achieving successful disposal of the spent fuel and preventing the need for additional processing. It is difficult to assess waste form acceptability for the melt and dilute process at this time because the acceptance criteria and requirements are still being defined. However, some aspects of the process would contribute to ensuring an acceptable waste form. For example, the melt and dilute process involves isotopic dilution of enriched uranium with depleted uranium. This reduces the criticality potential. Even so, preliminary criticality evaluations indicate that neutron poisons might have to be added to the spent fuel melt. Investigations are under way to explore the compatibility of neutron poison material with the fuel melt and with the canister.

Another possible advantage of the melt and dilute process is the relatively homogeneous waste form, which potentially eliminates the need for pretreatment characterization or a detailed fuel history. The ability to sample each fuel melt can provide a great deal of characterization information for ensuring that the waste form stays within the assumptions of the performance assessments. Unfortunately, specific characterization requirements are not clearly defined and could impose a significant post-treatment characterization burden on the process.

The melt and dilute process makes it possible to optimize the performance attributes (e.g., corrosion resistance, leachability) of the final waste form by controlling the relative amount of aluminum and uranium. This capability comes at the cost of having to adjust and confirm the alloy composition of each melter charge.

Several aspects of the melt and dilute process enhance its ability to produce a waste form acceptable for the repository. However, at present only borosilicate glass has been qualified for disposal in the federal repository. "One of the major challenges to the...disposition of DOE-owned Al-clad spent nuclear fuels is the lack of clear definition of the requirements for repository acceptance criteria of the fuels" (Westinghouse Savannah River Company, July 1998, p. 4.4).

6.2 Direct Codisposal

6.2.1 Risks Associated with Operating the Process

The direct codisposal process can be thought of simply in terms of dry handling of spent fuel. While this is an oversimplification, the process does not involve chemically altering the fuel or introducing energy sources. Thus, the process-related risks identified here are due primarily to handling.

Criticality. Storage of highly enriched spent fuel always poses the potential for criticality. As noted earlier, when the fuel is in wet storage, the consequences of an inadvertent criticality are substantially mitigated by the shielding and confinement provided by the water. Like the melt and dilute process, direct codisposal involves handling the fuel out of the basin water. Fortunately, the proposed location of the process in the process area of the L-Reactor facility would provide substantial shielding.

Contamination. Drying and the dry handling of spent fuel would contribute to contamination of the process areas. Handling and repackaging of damaged fuel would increase the contamination potential.

Direct Exposure. The direct codisposal process involves substantial handling of highly radioactive spent fuel out of the basin water. Current proposals to locate the process within the L-Reactor facility should result in adequate shielding for the remote operations. The transition to nonremote handling and packaging activities would provide an opportunity for direct radiation exposure.

6.2.2 Risk of Not Being Successfully Implemented

Design, Construction, and Startup. Of all three alternatives, direct codisposal appears to be the simplest technology. There are, of course, technical challenges associated with implementing a remotely operated spent fuel packaging process. Also, locating this process in the existing L-Reactor facility could pose additional challenges. However, the simplicity of drying and packaging suggests there is a reasonable probability of successfully designing, constructing, and operating this process.

Repository Acceptance Requirements. A large portion of the aluminum spent fuel is highly enriched and presents special criticality considerations. This criticality concern is aggravated by the more rapid corrosion and the lower burnup of research reactor fuel relative to commercial fuel, and hence a more rapid loss of subcritical geometry and potentially greater nuclear reactivity than is the case with commercial fuel. Studies to date indicate that without the addition of a neutron poison, there is a small but significant probability of postclosure criticality. The neutron poison would have to be chosen to provide the appropriate long-term dissolution rates as compared with the

aluminum spent fuel. One form of neutron poison being considered is gadolinium phosphate.

The large amount of characterization anticipated for the direct codisposal process represents a significant disadvantage. The relatively simple process of drying and packaging the fuel could be overwhelmed by the extensive characterization measurements required to meet repository acceptance criteria.

Unique Spent Fuel. Some of the spent fuel inventory, such as the powdered spent fuel and the significantly damaged spent fuel, would require additional treatment prior to direct codisposal. The repackaging of these spent fuels would likely be restricted by the repository waste acceptance criteria. Powdered fuel, for example, would probably need to be solidified. These additional treatment operations might be relatively simple, but they have not yet been developed.

Overall, there is a relatively large uncertainty involved in the ability of the direct codisposal process to produce an acceptable repository waste form.

6.3 Conventional Processing

6.3.1 Risks Associated with Operating the Process

The facility most likely be used to process the aluminum research reactor fuel under the conventional processing alternative is the H-Canyon. The F-Canyon facility could also be used, with some modifications, since the two facilities are very similar. The source terms involved in H-Canyon processing can be significant. In addition, there is some potential for energetic events involving flammable solvent, reactive chemicals, and hydrogen gas. The Basis for Interim Operations (BIO) for the H-Canyon addresses the potential health and safety risks (Westinghouse Savannah River Company, April 1998). The BIO evaluates the risk of routine operations and potential accident conditions. It identifies dominant accident scenarios, including loss of containment, transfer errors, uncontrolled chemical reactions and explosions, fires, criticality, inadvertent personnel exposure, and natural phenomenon events. Equipment and process designs, in addition to the canyon structure, serve to mitigate the risk of these postulated accidents to an acceptable level. The BIO also concludes that routine releases from H-Canyon operations are negligible.

The H-Canyon design has allowed successful operations for many years. In addition, years of operating experience have resulted in corrections, upgrades, and additions to the facility to address lessons learned. Significant events during the H-Canyon operating history are described in the BIO, as are the corrective actions taken to prevent these and other events.

Because of the many years of H-Canyon operations at SRS, there is some certainty in the calculated risk assigned to that facility. The facility has undergone several upgrades in safety analysis and in the determination of operational readiness.

6.3.2 Risk of Not Being Successfully Implemented

There is minimal risk regarding the success of conventional processing in achieving stabilization and disposal of the aluminum spent fuel. As discussed earlier, conventional processing is currently being used to stabilize and dispose of the aluminum defense-related spent nuclear fuel. Also, the DWPF is currently vitrifying high-level waste and producing glass canisters that are qualified for acceptance in the repository. One aspect of high-level waste processing—the treatment of the high-level waste salt—requires additional development to support the vitrification of all of the SRS high-level waste.

6.4 Cost Comparison

Recent cost studies considered several variations of the three processing alternatives (Westinghouse Savannah River Company, December 1997 and May 1998). The melt and dilute alternative is evaluated as utilizing both a new facility and a modification to the L-Reactor facility. The direct codisposal alternative is also evaluated in both of these ways. The conventional processing alternative is evaluated as utilizing the SRS canyons through 2010, and then taking one of four actions to accommodate the tail of fuel receipts from 2011 to 2035 (see Section 2.3): (1) build a new conventional processing facility, (2) build a new melt and dilute facility, (3) build a new direct codisposal facility, or (4) send the tail of fuel receipts to INEEL to utilize the dry transfer facility expected to be operational there. Some of the results from the cost study are provided in Table 2.

Table 2. Cost Comparison for Processing Alternatives

Alternative	10-Year Cost (FY98 millions)	39-Year Life Cycle Cost (FY98 millions)
Melt and Dilute at the L-Reactor Facility	\$916	\$1960
Direct Codisposal at the L-Reactor	910	1920
Conventional Processing Followed by Melt and Dilute at SRS	698	2060
Conventional Processing Followed by Direct Codisposal at SRS	702	2010
Conventional Processing Followed by New Conventional Processing Facility	704	2380

Source: Westinghouse Savannah River Company, May 1998, and U.S. Department of Energy December 1998a.

Table 2 shows that there is very little difference in cost between the direct codisposal and melt and dilute alternatives. The table also indicates that there is a minimal difference in the life-cycle cost (about \$100 million during 39 years) between the melt and dilute or direct codisposal alternative at the L-Reactor facility on the one hand and the conventional processing alternative through 2010 followed by a new treatment facility on the other hand. The one exception is the alternative that includes a new conventional processing facility, showing a difference of about \$400 million.

Looking at the 10-year cost, there is a much larger difference between conventional processing and either the melt and dilute or direct codisposal alternative (about \$210 million during a 10-year period). Thus if conventional processing were used to stabilize and dispose of the spent fuel, there would be an opportunity for significant future savings if the projected tail of fuel receipts did not occur or if the capability to treat and dispose of the fuel at INEEL became available. Furthermore, even if these opportunities do not develop, there is no cost penalty associated with the need to build a new treatment facility when conventional processing capability is no longer available.

The above cost estimates are based on an assumption that the SRS canyons will no longer be viable after 2010. Under the conventional processing alternative, the development of a new treatment facility to replace the canyon processing is assumed to begin in 2005 (i.e., a fiscal year 2005 line item project) so that the new facility will be operational in 2010. Thus the need for a new treatment facility to accommodate the tail of fuel receipts would have to be identified by 2005. Much more information about spent fuel receipts and the actual need for a new treatment facility would be available at that time.

7. SUMMARY AND CONCLUSIONS

Aluminum spent fuel from research reactors is unlike commercial spent nuclear fuel in that the aluminum fuel is less robust, more susceptible to corrosion, and of a much higher uranium-235 enrichment. This aluminum fuel is currently stored in wet storage basins at SRS. Improvements in water chemistry control and correction of other vulnerabilities associated with the wet storage of spent fuel at SRS have improved the storage conditions. However, corrosion behavior is difficult to predict, and some of the fuel has already undergone damage and corrosion at previous storage locations. There is no fundamental disagreement on the vulnerability of aluminum fuel in wet storage basins. This vulnerability prevents consideration of continued long-term wet storage as a viable alternative for managing this spent fuel.

DOE has committed to removing this aluminum research reactor fuel from wet storage and placing it in stable interim storage in a form suitable for ultimate disposal. DOE is preparing to select a processing alternative for this fuel. Three alternatives are being considered: melt and dilute, direct codisposal, and conventional processing.

The planning period for these processing alternatives is through the year 2035—about 40 years. The total amount of fuel to be processed during this period is uncertain. Estimates have been reduced by more than 25 percent during the last couple of years as a result of reactor shutdowns and the decision of some foreign countries not to return their research reactor fuel to SRS. More reductions are probable. Also, the tail of fuel receipts during at least one-half and as much as two-thirds of the planning period may be grossly overestimated, if it occurs at all.

DOE has sponsored several studies related to the selection of a processing alternative for stabilization and disposal of this spent fuel. In those studies that considered conventional processing, this alternative was assigned the highest rating. However, most of the DOE-sponsored studies excluded conventional processing from consideration.

Comparison of the estimated accident-induced latent cancer fatalities that could result from operation of each of the processing alternatives was found not to be very useful in discriminating among the processing alternatives. On the other hand, the two alternatives involving new technology or processes, especially the melt and dilute alternative, introduce new or unique hazards that must be understood and addressed. The melt and dilute process involves molten spent fuel, volatilized fission products, and the handling of high-level waste without the benefit of a canyon structure or underground waste facilities.

If one compares the risk or uncertainty involved in obtaining a waste form suitable for stable interim storage and acceptable for ultimate disposal, conventional processing is clearly favored. Conventional processing is a well-understood technology that is currently being conducted using the SRS canyon facilities. The direct codisposal and especially the

melt and dilute alternatives would require significant process development. In addition, the waste form resulting from conventional processing has been qualified for disposal in the federal geologic repository. The melt and dilute and especially the direct codisposal processes produce a waste form with some uncertainties regarding its acceptability in the repository.

The life-cycle costs of the processing alternatives, allowing for the uncertainty in predicting a 39-year life-cycle cost, are generally comparable. The 10-year costs are generally most favorable for the conventional processing alternative. Cost comparisons of the alternatives are based on receipt and processing of spent fuel through 2035. The tail end of this inventory is highly uncertain. In addition, with accelerated receipts, the backlog of spent fuel could be eliminated as early as 2010. These uncertainties in the projected fuel inventories add to the uncertainty of the cost estimates for the alternatives.

Evaluation of the spent fuel inventory and the processing alternatives being considered leads to the conclusion that DOE ought to use conventional processing for aluminum research reactor spent fuel to the maximum extent practical. At a minimum, the SRS canyons ought to be used through 2010, allowing DOE to defer the decision on a replacement technology until at least 2005, when more information will be available on the type and number of future fuel receipts.

REFERENCES

Defense Nuclear Facilities Safety Board, *Staff Trip Report: Spent Fuel Storage: Status of DNFSB Staff Review of Spent Fuel Storage in the Basins at Savannah River Site*, Washington, D.C., June 3, 1993.

Defense Nuclear Facilities Safety Board, *Staff Trip Report: Savannah River Site Spent Fuel Storage Trip Report—January 27, 1994*, Washington, D.C., February 18, 1994.

Defense Nuclear Facilities Safety Board, *Recommendation 94-1, Improved Schedule for Remediation in Defense Nuclear Facilities Complex*, transmitted by letter from J. T. Conway to H. R. O'Leary, Washington, D.C., May 26, 1994.

Defense Nuclear Facilities Safety Board, *Staff Trip Report: Savannah River Site Spent Fuel Storage Trip Report—January 3–6, 1995*, Washington, D.C., January 23, 1995.

Defense Nuclear Facilities Safety Board, *Staff Trip Report: Spent Nuclear Fuel at the Savannah River Site (SRS)*, transmitted by letter from J. T. Conway to T. P. Grumbly, Washington, D.C., August 3, 1995.

Defense Nuclear Facilities Safety Board, *Stabilization of Deteriorating Mark 16 and Mark 22 Aluminum-Alloy Spent Nuclear Fuel at the Savannah River Site*, DNFSB/TECH-7, Washington, D.C., November 1, 1995.

Defense Nuclear Facilities Safety Board, letter from J. T. Conway to H. R. O'Leary, Washington, D.C., November 15, 1995.

Idaho National Engineering Laboratory, *Spent Fuel Background Report*, EGG-WM-11249, Great Falls, ID, March 1994.

National Research Council, *Research Reactor Aluminum Spent Fuel: Treatment Options for Disposal*, Washington, D.C., 1998.

Nuclear Regulatory Commission, *Review of the Aluminum-Based Research Reactor Spent Nuclear Fuel Disposition Program*, transmitted by letter to the Savannah River Operations Office, Washington, D.C., June 5, 1998.

Sandia National Laboratories, *A Multi-Attribute Utility Decision Analysis for Treatment Alternatives for the DOE/SRS Aluminum-Based Spent Nuclear Fuel*, Draft, May 23, 1998.

U.S. Department of Energy, *ACTION: A Decision on Phaseout of Reprocessing at the Savannah River Site (SRS) and the Idaho National Engineering Laboratory (INEL) Is*

Required, transmitted by memo from R. A. Claytor to Secretary of Energy James Watkins, Washington, D.C., April 28, 1992.

U.S. Department of Energy, *Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear Fuel and Other Reactor Irradiated Nuclear Materials and Their Environmental, Safety and Health Vulnerabilities*, Washington, D.C., November 1993.

U.S. Department of Energy, *Defense Nuclear Facilities Safety Board Recommendation 94-1 Implementation Plan*, transmitted by letter from Secretary of Energy H. R. O'Leary to J. T. Conway, Washington, D.C., February 28, 1995.

U.S. Department of Energy, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*, DOE/EIS-0203F, Idaho Operations Office, Idaho Falls, ID, April 1995.

U.S. Department of Energy, *Record of Decision for the Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (60 FR 28680), May 30, 1995

U.S. Department of Energy, *Final Environmental Impact Statement, Interim Management of Nuclear Materials*, DOE/EIS-0220, Savannah River Site, Aiken, SC, October 20, 1995.

U.S. Department of Energy, *Facility Utilization Strategy for the Savannah River Site Chemical Separation Facilities*, Washington, D.C., December 1995.

U.S. Department of Energy, *Record of Decision for the Interim Management of Nuclear Materials EIS* (60 FR 65300), Washington, D.C., December 19, 1995.

U.S. Department of Energy, *Final Environmental Impact Statement on a Proposed Nuclear Weapons Non-proliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel*, DOE/EIS-0218F, Washington, D.C., February 1996.

U.S. Department of Energy, *Record of Decision for the Interim Management of Nuclear Materials EIS* (61 FR 6633), February 21, 1996

U.S. Department of Energy, *Technical Strategy for the Treatment, Packaging, and Disposal of Aluminum-Based Spent Nuclear Fuel, A Report of the Research Reactor Spent Nuclear Fuel Task Team*, Draft, Washington, D.C., May 1996.

U.S. Department of Energy, *Record of Decision on a Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (61 FR 25092), Washington, D.C., May 13, 1996.

U.S. Department of Energy, *Technical Strategy for the Treatment, Packaging, and Disposal of Aluminum-Based Spent Nuclear Fuel, A Report of the Research Reactor Spent Nuclear Fuel Task Team*, Washington, D.C., June 1996.

U.S. Department of Energy, *Quality Assurance Requirements and Description for the Civilian Radioactive Waste Management Program, Revision 6*, DOE/RW-0333P, Washington, D.C., March 3, 1997.

U.S. Department of Energy, *Record of Decision for the Interim Management of Nuclear Materials EIS* (62 FR 17790), Washington, D.C., April 11, 1997.

U.S. Department of Energy, *Evaluation of Codisposal Viability for Aluminum-Clad DOE-Owned Spent Fuel: Phase I Intact Codisposal Canister*, BBA000000-01717-5705-00011 REV 0, Yucca Mountain Site Characterization Project Office, Las Vegas, NV, June 9, 1997.

U.S. Department of Energy, *Savannah River Site Chemical Separation Facilities Multi-Year Plan*, Washington, D.C., July 1997.

U.S. Department of Energy, *Revised Expected Receipts of Foreign Research Reactor Spent Nuclear Fuel at Savannah River Site*, transmitted by letter from K. A. Chacey (DOE Office of Spent Fuel Management) to L. Watkins (DOE-SR), Washington, D.C., July 2, 1997.

U.S. Department of Energy, *Foreign Research Reactor Fuel Acceptance Criteria—Failed Fuel Report*, DOE/SNF/REP-013, Washington, D.C., November 1997.

U.S. Department of Energy, *Nonproliferation Study for Foreign Research Reactor Spent Nuclear Fuel*, letter from D. G. Huizenga (DOE-EM) to C. Fitzgerald (DOE-NN), Washington, D.C., April 15, 1998.

U.S. Department of Energy, *Nuclear Materials Processing Needs Assessment*, presentation by A. Guevara and G. Klipa, Washington, D.C., May 6, 1998.

U.S. Department of Energy, *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement*, DOE/EIS-0279D, Aiken, SC, December 1998.

U.S. Department of Energy, *Nonproliferation Impacts Assessment for the Management of the Savannah River Site Aluminum-Based Spent Nuclear Fuel*, Office of Arms Control and Nonproliferation, Washington, D.C., December 1998a.

U.S. Department of Energy, *Report on the Savannah River Site Aluminum-Based Spent Nuclear Fuel Alternatives Cost Study*, Savannah River Operations Office, Aiken, SC, December 1998b.

U.S. Department of Energy, *Spent Fuel Database (SFD)*, Version 3.3.2, DOE Department of Spent Fuel Management, Washington, D.C.

Westinghouse Safety Management Solutions, Inc., *Safety Evaluation of the TSS Melt and Dilute Process in 105-L (U)*, WSMS-LIC-98-0011, Aiken, S.C., February 26, 1998.

Westinghouse Savannah River Company, *Canyon and Reactor Disassembly Basin Production Rates for SRS and FRR Spent Nuclear Fuels (U)*, NMP-PLS-950159, Aiken, SC, May 25, 1995.

Westinghouse Savannah River Company, *Chemical Stabilization of Defense Related and Commercial Spent Nuclear Fuel at the Savannah River Site (U)*, NMS-PLS-950239, Aiken, SC, August 16, 1995.

Westinghouse Savannah River Company, *Savannah River Site Evaluation of Spent Nuclear Fuel Options (U)*, WSRC-RP-95-798, Aiken, SC, February 1996.

Westinghouse Savannah River Company, *Acceptance Criteria for Interim Dry Storage of Aluminum-Alloy Clad Spent Nuclear Fuels (U)*, WSRC-TR-95-0347, Aiken, SC, March 1996.

Westinghouse Savannah River Company, *The Corrosion of Aluminum-Clad Spent Nuclear Fuel in Wet Basin Storage*, WSRC-MS-96-0141, Aiken, SC, June 16, 1996.

Westinghouse Savannah River Company, *Decision Factors for Disposition of DOE-Owned Aluminum Clad Spent Nuclear Fuel at the Savannah River Site*, WSRC-MS-96-00154, Aiken, SC, June 17, 1996.

Westinghouse Savannah River Company, *Spent Nuclear Fuel EIS Routine Release Environmental Dosimetry Calculations*, SRT-ETS-960149, Aiken, SC, December 20, 1996

Westinghouse Savannah River Company, *Savannah River Site Spent Nuclear Fuel Environmental Impact Statement Engineering Data Book for Routine Releases*, WSRC-TR-97-0044 Rev 1, Aiken, SC, March 25, 1997.

Westinghouse Savannah River Company, *Alternative Aluminum Spent Nuclear Fuel Treatment Technology Development Status Report (U)*, WSRC-TR-97-0084, Aiken, SC, April 1997.

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Westinghouse Savannah River Company, *Spent Nuclear Fuel Alternative Technology Decision Analysis (U)*, U-ESR-G-00004, Aiken, SC, June 26, 1998b.

Westinghouse Savannah River Company, *SNF Alternate Technology Disposition Recommendation*, transmitted by letter from W. G. Poulson to J. E. Anderson, Aiken, SC, June 26, 1998b.

Westinghouse Savannah River Company, *Basis for Functional Performance Requirements for a Spent Nuclear Fuel Treatment and Storage Facility*, WSRC-TR-98-00228, Aiken, SC, July 1998.

Westinghouse Savannah River Company, *Spent Nuclear Fuel Alternative Technology Risk Assessment (U)*, Y-TRA-G-00001, Aiken, SC, July 16, 1998.

Westinghouse Savannah River Company, *Savannah River Site FY 1999 Spent Nuclear Fuel Interim Management Plan*, WSRC-RP-98-00713, Aiken, SC, November 1998.

Westinghouse Savannah River Company, *Spent Nuclear Fuel—Treatment and Storage Facility Project Risk Analysis Report (U)*, Y-RAR-L-00001, Aiken SC, December 14, 1998.

ABBREVIATIONS AND ACRONYMS

Abbreviation	Definition
BIO	Basis for Interim Operations
Board	Defense Nuclear Facilities Safety Board
DOE	Department of Energy
DWPF	Defense Waste Processing Facility
EIS	Environmental Impact Statement
FRR	Foreign Research Reactor
HEPA	high-efficiency particulate air
HEU	highly enriched uranium
INEEL	Idaho National Engineering and Environmental Laboratory
RBOF	Receiving Basin for Offsite Fuel
ROD	Record of Decision
SNF	Spent Nuclear Fuel
SRS	Savannah River Site
WSRC	Westinghouse Savannah River Company

DOE Response to Comment Letter L-17

Defense Nuclear Facilities Safety Board

June 8, 1999

The Honorable John T. Conway
Chairman
Defense Nuclear Facilities Safety Board
625 Indiana Avenue, NW, Suite 700
Washington, D.C. 20004-2901

Dear Mr. Chairman:

Thank you for the comments you provided in your letter dated June 8, 1999, and the accompanying Technical Report regarding the Savannah River Site (SRS) aluminum research reactor spent nuclear fuel disposition. We understand the importance of the issues you raised concerning deployment of new technology for preparing aluminum spent fuel for disposal. We also recognize that the Board, and others, favor using the SRS canyons for this fuel because it is a proven technology. However, the canyons are fully committed for the next several years and the Department believes it is prudent to begin planning now for the eventual replacement of the aging SRS canyons with a technology and facility which are more cost-effective over the long term for the preparation of spent fuel for ultimate disposal.

Please be assured that the Department is proceeding along a calculated path of research and development, testing, facility/equipment design, construction, and operations which will lead to safe implementation of an alternate technology. The Department is committed to building on the progress already made in the laboratory development of the melt and dilute process by validating the safety, viability, and cost of deployment of the technology through pilot-scale demonstration and analysis before starting a capital project. To ensure availability of the capability to prepare fuel for disposal, and if necessary stabilize fuel for health and safety reasons, SRS Canyon processing capability will be retained until we have successfully demonstrated implementation of the alternate treatment technology.

The specific concerns raised in your report are being reviewed in the context of our on-going National Environmental Policy Act (NEPA) evaluation of processing alternatives for the Savannah River Site. Specific responses to comments in the Technical Report that pertain to our NEPA impacts analysis will be addressed in the final environmental impact statement which will be completed later this year.

As always, we will continue to work closely with the Board and staff. We are prepared to meet with you to review the current status of this program and discuss the technical aspects in more detail.

Sincerely,

Carolyn L. Huntoon
Assistant Secretary for
Environmental Management

**DOE RESPONSE TO DEFENSE NUCLEAR FACILITIES SAFETY BOARD
TECHNICAL REPORT DNFSB/TECH-22
SAVANNAH RIVER SITE SPENT NUCLEAR FUEL**

The purpose of this document is to provide responses to the major issues raised by the Defense Nuclear Facilities Safety Board (DNFSB) staff regarding planned actions proposed by the Department of Energy (DOE) for the treatment and storage of aluminum research reactor spent nuclear fuel at the Savannah River Site (SRS). The DNFSB Technical Report was transmitted to James M. Owendoff, then Acting Assistant Secretary for Environmental Management, from John T. Conway, Chairman, DNFSB, on June 8, 1999.

The DNFSB staff concerns, observations, and comments focused on the Alternate Technology Program for the development of the melt and dilute technology for treatment of aluminum research reactor fuel in preparation for onsite storage and eventual disposal in a geologic repository. Our response addresses issues associated with continued wet storage of spent fuel (Section 3 of the DNFSB Technical Report), risk estimates (Section 6), the risks associated with operating the melt and dilute process (Section 6.1.1), and the risks of not successfully implementing the process (Section 6.1.2).

INTERIM WET STORAGE OF SPENT FUEL

The issue is that corrosion behavior of spent fuel stored in wet basins is difficult to predict, and some of the fuel has already undergone damage and corrosion at previous storage locations.

The vast majority of research reactor fuel in storage at SRS is in excellent condition with respect to corrosion. The fuel with signs of corrosion initiation readily passivates under clean water conditions. The vulnerabilities and the potential for release from the worst of the foreign aluminum alloy based research reactor fuel identified to date has been found to be so insignificant that the Nuclear Regulatory Commission recently granted cask certificates to certain cask vendors to transport the fuel without any canning. By contrast, the Taiwanese Research Reactor fuel, posed a greater risk because the fuel was severely damaged, stored in leaky cans, and had a uranium metal core. However, this fuel has already been processed in the SRS canyon processing facilities.

The SRS Receiving Basin for Offsite Fuel (RBOF) and L-Area basin maintain excellent water chemistry and fuel storage practices to minimize corrosion. The present water qualities of both RBOF and L-Basin are excellent as measured, in part, by typical conductivities of <1 and <10 μ Siemens per centimeter, respectively. RBOF, which has maintained good water chemistry since its inception, has had no instances of corrosion of aluminum-clad SNF with an aluminum-uranium core that was stored for over 20 years. A 1997 project to upgrade the L-Basin included basin vacuuming, monitoring wells, basin level detection system, and continuous circulation through mixed-bed deionizers. Furthermore, galvanic couples (e.g., stainless steel and aluminum) in the basins, which along with poor water chemistry caused corrosion initiation in the defense production related aluminum SNF, have been eliminated. No corrosion of aluminum has been seen in the disassembly basins since the upgrade.

Safe storage of the aluminum-based fuel, including fuel with breached cladding, can be maintained for at least 10 to 20 years in the SRS basins under present conditions. This conclusion is based on site experience and supported by the results from site surveillance programs and laboratory and field test programs. The expected impact of basin storage of fuels with minor cladding breaches is an increase in the basin water activity that is well within the envelope of safe operation of the SRS basins. A recent validation of the SRS aluminum basin management practices was manifested in the International Atomic Energy Agency's decision to use the SRS basin water chemistry and corrosion prevention strategy as the basis for a standard for the management of aluminum spent nuclear fuel basins around the world.

RISK ESTIMATES

The issue is that the risk estimates associated with operation of the melt-dilute facility that were reported in the draft SRS Spent Nuclear Fuel Management Environmental Impact Statement represent mitigated risk, and so include significant assumptions that may be inappropriate.

Appendix D of the EIS now shows mitigated and unmitigated impacts for new technologies where the accident frequencies warranted such treatment. Analyses have been performed based on realistic, qualitative assumptions about the melt and dilute process. Assumptions are based on experimental results from laboratory melter testing, experience from actual melters, analyses supporting the L-Area Experimental Facility (LEF, pilot test facility for the melt and dilute system), and industry-based studies of Uranium-Aluminum melting operations.

The actual design of the full-scale melt and dilute system has not been completed. More rigorous analyses will have to be conducted as part of the Safety Analysis Report development during the design and construction phase of the melt and dilute system. Safety systems will be identified and assumed to be present during melter operations. It is during this stage of development where the analytical assumptions used to support the EIS will be identified to be correct or incorrect. At that stage (if warranted by the design output document and the Safety Analysis Reports), other mitigative steps can be taken to maintain analyzed accident releases below the regulatory requirements and other acceptable boundaries.

SAFETY ISSUES ASSOCIATED WITH OPERATING THE PROCESS

Melter Off-Gas: The issue is that the melt and dilute process would require an off-gas treatment system to remove volatile radioisotopes in the gases and vapors that would result from melting the fuel.

A critical technology element in the development of the melt-dilute process is the development of an off-gas system. The volatilization of radioactive species during the melting stage of the process primarily constitutes the off-gas in this process. Some of the key species that have previously been studied are iodine and cesium. These species have been shown to volatilize during melting experiments; however, the degree to which they are released is highly dependent upon atmosphere, fuel burn up, temperature, and fuel composition. With this in mind, an analytical and experimental program has been undertaken at SRS to assess the volatility and capture of species under the melt-dilute operating conditions.

The analytical and experimental tests conducted at both the Savannah River Technology Center (SRTC) and Argonne National Laboratory have identified cesium as the melt constituent of most concern with respect to volatilization. Experimental tests using both cesium surrogates and radioactive cesium have shown that zeolite is an effective cesium trap. As a result, a preliminary off-gas system concept has been developed employing dry zeolite absorber beds as the primary cesium trapping medium. Validation of this off-gas concept will continue to occur during the bench-scale tests in hot cells and during the pilot-scale testing of irradiated fuel in the LEF.

TC The effectiveness of zeolite (or molecular sieve filter) as an absorption bed for cesium and other radionuclides has been well demonstrated in the laboratory at SRTC. Zeolite has also been used as the filter for capturing the cesium in the molten salt in the electrometallurgical process being demonstrated at Argonne National Laboratory – West. Studies in the 1960's (Wolkoff, J. and Chilenskas, A., "Melt Refining of Irradiated Uranium," Nuclear Science and Engineering, 9, pp 71-77, 1961) showed that molecular sieves can retain cesium at high loads. More recent studies (Pereira, C. et al, "Overview of Mineral Waste Form Development for the Electrometallurgical Treatment of Spent Nuclear Fuel," Proceedings of the Third Topical Meeting on DOE Spent Nuclear Fuel and Fissile Material Management, Charleston, SC, 1998) of the electrometallurgical process also demonstrated that fission products in the molten salt are effectively exchanged from the salt into the zeolite structure.

Stirring requirements, uranium sampling methods, and dilution techniques have been successfully demonstrated at the planned operating temperature of 850°C. The impact of stirring the melt on fission product release has been studied using surrogate materials at SRTC. Bench-scale surrogate volatility tests were conducted using a resistance furnace with no stirring. Full-size fuel surrogate melts using induction stirring show only an incremental increase in release amounts. The relative release fractions from these two tests fell within the same order of magnitude.

The off-gas system for the melt-dilute treatment technology will be designed with the appropriate engineering controls such as are necessary to handle both normal and off-normal events during processing. The LEF will be designed under the same constraints and will be used to validate this engineering approach to prevent the unfiltered release of radioactive material. For example, the confinement enclosure for the LEF will meet seismic requirements and the back-up power supply, automatic shut off, and other redundant safety features will be designed and built with defense-in-depth to the latest codes and standards.

Building Ventilation and Confinement: *The issue is that the existing L-Reactor process area ventilation system would not provide the reliable confinement function needed for melt and dilute operations.*

A new confinement system and structure would be designed to the latest codes and standards which will compensate for any limitations of the existing confinement capability of the L-Reactor facility.

Steam Explosions: *The issue is that the introduction of water in or around the melter would result in the potential for an energetic steam explosion event.*

The processing procedures include a drying stage whereby the fuel assemblies will be furnace dried to remove any free water. Experimental activity is underway to demonstrate the drying strategy. Although in actual operation, the spent nuclear fuel would be furnace dried, tests have been performed using mock fuel assemblies which were placed under water at SRTC, drip dried, and then put through the melt-dilute cycle with no adverse consequences.

TC Additionally, engineering controls have been designed into the furnace to prevent the possibility for melt-water interactions from the cooling water supply of the induction furnace. A quartz furnace liner has been installed in the developmental furnace used at SRTC at a position such that it acts as a permanent physical barrier preventing any water from a leaking furnace cooling line from contacting the outer wall or internal cavity of the crucible. Furthermore, redundant moisture sensors have been installed in the furnace lining that effects immediate shut down of furnace power if alarmed. These engineering controls, as well as redundant sensors and controls, are included in the LEF design. Although melt-water interaction during melting operations can be serious, induction furnace systems are the mainstay of the DOE and commercial foundry industry both nationally and internationally. Both Savannah River Site and Oak Ridge National Laboratory safely and efficiently operated production-scale induction furnaces for more than 30 years during the manufacturing of defense related uranium-aluminum fuel assemblies. The appropriate engineering controls and procedures were in place then and would be implemented for this project.

Criticality: *The issue is that storage and treatment of highly enriched spent nuclear fuel creates the potential for criticality.*

Detailed safety analyses have been performed to determine the probability of occurrence of a criticality event in the Treatment and Storage Facility (TSF) given the current pre-conceptual design configuration. These analyses have shown that with the proper engineering controls and procedures, the probability of a criticality event is extremely unlikely ($<10^{-6}$).

Calculations have been performed as part of the conceptual design effort for the TSF regarding the maximum allowable concentration of uranium-235 permitted in the induction furnace per melt batch. Disciplined operations that are well established at the SRS along with strict engineering controls and procedures will preclude the overloading of the furnace during operation. For example, the M-Area Reactor Materials Program consistently and safely produced uranium-aluminum fuels for the SRS for over 30 years. These activities involved the melting of uranium-aluminum alloys comprised of large percentages of fissile uranium-235 on the order of 93% enriched.

Contamination: *The issue is that continually increasing contamination, poor equipment performance, and the need for personnel access to maintain the equipment could present significant operational challenges.*

The entire melt-dilute system would be located within a confinement enclosure to minimize facility contamination. In addition, rigorous handling procedures and controls would be implemented to reduce the spread of contamination. Methods to accomplish this objective are already being developed. For example, the addition of dilution/compositional adjustment materials to the melt during treatment would be performed while the furnace is cold prior to heating. The off-gas system would then be attached and the entire system sealed. This seal would not be broken until the melt batch is complete and cool. Following the completion of the treatment cycle, removal of the ingot would be performed using remote handling operations. To minimize the spread of contamination during these operations, the crucible-liner melting system is being developed such that the furnace liner is integral to the ingot and thus provides containment of any particulate contamination during handling. Also, demonstration of the risk mitigation strategies to prevent gross facility contamination along with the necessary remote handling operations would be performed in the LEF.

Waste Handling: *The issue is that the condensing medium in the melter off-gas system would need to be removed periodically for disposal.*

The melt-dilute process waste stream is currently being developed on a dry waste stream basis with the liquid waste stream option as a backup. The dry waste stream option has three potential alternatives for the disposal of process wastes such as the cesium-loaded zeolite absorber beds. The first alternative is to place the absorber beds into the DOE spent nuclear fuel canisters along with the melt-dilute ingots for transportation and disposal at a geologic repository. The second alternative is to simply dry store the absorber beds in canisters in the TSF along with the spent fuel canisters. The half-life of cesium-137 is only 30 years and following 1 to 2 half-lives the materials could be shipped to the SRS E-Area for disposal as solid low-level waste. The third alternative involves placing the zeolite absorber beds into an appropriate high-level waste tank, if one were available, where they would be vitrified along with the other waste in the tank.

Furnace Material Carryover: *The issue is that a small fraction of the uranium and plutonium content of each melt batch may escape from the liquid surface and eventually adsorb on the filters or elsewhere in the off-gas system or ventilation system as metallic and oxide dust particles or fines. This possibility must be addressed relative to criticality and material release.*

Based on lessons learned from SRS experience, we anticipate that there will be multiple barriers/filters in the melt-dilute system, so releases to elsewhere in the off-gas system would be unlikely. Detailed characterization of the off-gas particulates entrapped in the filter in the SRTC melt-dilute system is being performed and will be validated in the LEF. The Department does not expect that the issue will pose a significant safety concern.

With regard to past aluminum-uranium casting operations at SRS during which uranium-235 was deposited within the ventilation ductwork (i.e., in the M-area reactor materials production area), the operations were quite different than what is proposed for the TSF. The melting operations in M-area used many different types of materials in their melt batches. In addition to large chunks of enriched uranium, they routinely handled and melted lathe chips and other small uranium fines which most definitely contributed to the materials deposited in the exhaust ductwork. Additionally, all of the fuel fabrication lathe milling

machines utilized exhaust hoods that fed into the central furnace ducting system. Historical records from the 321-M building show that while the exhaust hoods directly above the casting furnaces did show evidence of uranium particulate, the greater concentrations of particulate uranium materials were found in the various other mechanical devices such as the lathes, chip compactor, and log saw.

SRS possesses extensive experience in the manufacturing of cast uranium-aluminum fuels. While there are some differences between fuel fabrication and irradiated fuel treatment, the basic casting/foundry operations are the same. Furthermore, experience exists within the DOE complex concerning the melting of irradiated fuel. Experimental Breeder Reactor spent fuel was processed from 1965-69 at the Fuel Conditioning Facility at Argonne National Laboratory- West (ANL-W) using the Melt-Refining Operation.

Unique Spent Fuel: *The issue is that powdered spent fuel and fuel stored in various configurations within canisters and storage tubes represent unique challenges.*

TC The Department has identified conventional processing to manage the powdered spent fuel currently in inventory at SRS. However, most of the powdered fuel inventory projected for receipt at SRS does not currently exist. It would thus seem feasible that interaction with the eventual producer could help reduce the potential need for repackaging; i.e., we could provide a ventable aluminum can design or at least suggest the use of aluminum cans to facilitate handling and treatment using the melt-dilute process.

RISK OF NOT BEING SUCCESSFULLY IMPLEMENTED

Design, Construction, and Startup: *The issue is that the melt and dilute process requires significant technology development such as an off-gas system and remote handling. Also, the shortcomings of the L-Reactor facility would have to be addressed in the design of a more elaborate, self-contained melter.*

Although the L-reactor building was not originally designed to house the melt-dilute process, upgrades/modifications to the existing reactor process room ventilation and exhaust system to achieve the necessary level of confinement would be made. Furthermore, the current design of the off-gas system for both the L-Area Experimental Facility (LEF) and Treatment and Storage Facility (TSF) provides self-confinement. The design is such that the furnace is inside a containment box that is at negative pressure with respect to the ambient environment. Successful operation of the LEF furnace and off-gas system would provide validation of this self-confined/contained furnace off-gas concept for the TSF.

The SRS has extensive experience with remote operations of processes related to the handling of radioactive materials. Several existing site missions, including the Defense Waste Processing Facility and the Canyons, involve extensive use of remote handling operations for day-to-day operations. The development of the appropriate remote handling and operations procedures has been integral to the melt-dilute program. The SRTC LEF simulator and the LEF would demonstrate remote melt sampling, induction stirring, ingot removal, and other spent fuel handling operations.

Experiments conducted at SRTC have resulted in the adoption of a crucible-liner system comprised of a carbon steel liner with a graphite crucible. Graphite crucibles are widely used in induction furnaces throughout the casting industry including at SRS. Graphite is an excellent crucible material providing good temperature uniformity and heat conduction. In tests conducted at SRTC using the developmental furnace, graphite crucibles have been used for approximately 100 melts without failure. The use of a carbon steel liner has been demonstrated routinely without failure at SRTC. While carbon steel does react with the uranium-aluminum melt, the interaction below 1000°C is very limited.

Remote handling operation of this crucible-liner system is being designed. Replacement of the crucible-liner system could be integrated into the single step of furnace charging of the spent fuel assemblies; i.e., the spent fuel in an aluminum basket would be placed inside the crucible-liner system and the entire package loaded into the furnace simultaneously in a single remote operation.

Achievement of the desired alloy composition would be performed during the fuel loading operation as the dilution and compositional adjustment materials will be integral to the basket containing the spent fuel assemblies. The required amounts of dilution materials can be calculated using available data. The spent fuel form composition window in the repository license application should be sufficiently broad so that determining the exact composition is not required, but rather the process would require only that the composition be determined within some relatively broad limits.

TC

The bridging phenomenon is only of concern when the furnace charge materials do not occupy a significant volume of the crucible and when no "mechanical" constraint is imposed on the charge. For the melt-dilute process, the spent fuel assemblies would be transported to the furnace in an aluminum basket, which would be melted along with the fuel, and would provide rigid mechanical support to the assemblies thus preventing bridging.

In summary, DOE believes that it can meet technology development challenges and that the risk that the Department could not implement the melt and dilute process is small (assuming that the out year budgets support project completion.) Nevertheless, DOE intends to maintain canyon processing capability until an alternative technology is successfully demonstrated.

Repository Acceptance Requirements: *The issue is that it is difficult to assess waste form acceptability because the acceptance criteria and requirements are still being defined. Hence, evolving requirements may impact repository acceptability of the melt-dilute disposal form.*

Preliminary requirements for acceptance of repository disposal forms are contained in Revision 3 of the Waste Acceptance System Requirements Document (WASRD), DOE/RW-0351. Issued in April 1999, this document reflects repository design and analysis concepts as captured in the Reference Design included in the December 1998 Viability Assessment. Based upon the performance analysis results from the Viability Assessment, changes have since been made to the Reference Design. These changes, as well as findings from other concurrent analyses and studies, will need to be reviewed and the WASRD revised accordingly. However, the revisions to the WASRD (due to the new Reference Design or its potential subsequent evolution) are expected to have little, if any, impact on the melt-dilute disposal form as the design changes were made to enhance overall repository performance and were not specific to the disposal canister design that is to be used for the melt-dilute product.

TC

Although the criteria will not become final until a repository is licensed for operation, we expect that in the next year the criteria will become sufficiently firm such that further development of the melt-dilute technology can proceed with minimal risk. The results from the 1998 Viability Assessment show that the contribution to the overall repository performance for all of the DOE managed spent fuel is less than that for a comparable amount of commercial spent fuel. The melt-dilute disposal form presents even less of a challenge to repository performance than the N-reactor fuel which was used as the representative fuel in the Viability Assessment analyses.

Additionally, in their pre-licensing reviews of DOE spent fuel, the Nuclear Regulatory Commission staff have indicated that the melt-dilute form would be an acceptable concept to consider for ultimate disposal. For all of these reasons, DOE does not believe that uncertainties about repository acceptance requirements should be a factor in the Department's decision making regarding proposed melt and dilute treatment. The Final EIS addresses this issue in Section 2.2.1.

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Comment Letter L-18
Citizens for Nuclear Technology Awareness
July 12, 1999



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July 14, 1999

Honorable William B. Richardson
Secretary of Energy
U.S. Department of Energy
1000 Independence Avenue SW
Washington, DC 20585

991649

(RIG/SENU)

The selection of a processing alternative, Mr. Secretary, . . .

. . . for the disposal of aluminum-clad spent fuel from research reactors has a particular interest for Citizens for Nuclear Technology Awareness (CNTA) because whichever alternative is selected, the processing will take place at the Savannah River Site.

CNTA is the largest grass-roots organization in the United States that advocates full utilization of the beneficial applications of nuclear energy. Our 1600 members, who live primarily in Georgia and South Carolina, include a substantial number of nuclear scientists and engineers. One of our primary functions is to debunk nuclear fables and myths by providing factual information, and that's the purpose of this letter.

You recently received a letter from the Nuclear Control Institute, Natural Resources Defense Council, and Institute for Science and International Security which characterized reprocessing of this fuel in the SRS "canyons" as expensive, creating great volumes of liquid waste, and producing weapons-usable nuclear materials. We strongly disagree, and we are concerned that such technically incorrect statements could lead DOE to ill-advised decisions.

CNTA endorses the recent conclusion of the Defense Nuclear Facilities Safety Board. Specifically, DOE should maximize the use of its existing processing facilities at SRS, at least to the year 2010, to dispose of this nuclear fuel. Furthermore, we are concerned, as the DNFSB points out in its letter of June 8, 1999, that the melt and dilute alternative process does not appear to be based on a comprehensive consideration of safety. The DNFSB was created to ensure greater safety of the public in the vicinity of SRS and other nuclear defense facilities. The members of our organization would be very troubled if DOE disregarded the Board's advice regarding safety.

Nuclear experts in CNTA, whose number and professional credentials most likely far exceed those in the NCI, NRDC and ISIS, urge you to consider the following in selecting the processing alternative:

- Reprocessing is a proven technology, currently in use both at the Savannah River Site and worldwide. Furthermore, both the National Research Council and the Defense Nuclear Facilities Safety Board have recommended reprocessing of aluminum-clad fuel until the DOE has a demonstrated capability (i.e., a working facility) for disposition. In fact, the DNFSB, in a June 8 letter to the Department, characterized the SRS processing canyons as "workhorses," and encouraged DOE to "protect and capitalize on its existing capability where safety assurance has already been demonstrated rather than hastening to replace it."
- Studies by DOE and others have shown that the cost of reprocessing fuels at SRS is competitive with alternative methods, including melt and dilute. In fact, if a reasonable credit is taken for the value of the low-enriched uranium, which could be used to produce electricity, then reprocessing is cheaper. We believe DOE's experience with Hanford's N Reactor spent fuel is indicative of the costs and continuing uncertainties that have resulted from foreclosing the reprocessing option.
- How much waste will eventually be disposed of in a geologic repository? We believe that is a fundamental question that should be asked in evaluating any alternative for fuel disposition. Again, the National Research Council (among others) has confirmed that the volume of borosilicate glass going to the national repository from reprocessing would be less than the volume of high level waste going to the repository from the melt and dilute option.
- Reprocessing highly enriched research reactor fuels is intended to produce a low-enriched uranium product, not a weapons capable material. This isotopic dilution (to low-enriched uranium) eliminates for all time any potential for using this uranium for weapons. There is no concern with plutonium, because there is only a trace of plutonium in highly enriched spent fuel, and its isotopic composition is not suitable for making weapons. The Department correctly concluded in its Draft Environmental Impact Statement that reprocessing could be achieved in a way that is consistent with the nation's non-proliferation goals.

CNTA favors safe, rapid disposition of this spent nuclear fuel. Reprocessing is both safe and rapid, and is the logical way for DOE to meet its responsibilities to manage the research reactor fuel. Any other course of action would represent a substantial increase in the time for disposition of these materials, and a corresponding cost to the Department and the taxpayer. In addition, we, like the DNFSB, are concerned about the safety of speculative technology.

We appreciate your consideration of our views on this most important matter.

Sincerely,



Fred C. Davison
Chairman

cc: South Carolina Congressional Delegation
Senator Paul Coverdell
Senator Max Cleland
Representative Charlie Norwood
John Conway
T.J. Glauthier
Ernest Moniz
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DOE Response to Comment Letter L-18



Department of Energy

Washington, DC 20585

September 24, 1999

Dr. Fred C. Davison
Chairman
Citizens for Nuclear Technology Awareness
2711 Middleburg Drive
Suite 212
Columbia, South Carolina 29204

Dear Dr. Davison:

I have been asked to respond to your July 14, 1999 letter to the Secretary of Energy regarding the Department's pending selection of a treatment method for aluminum research reactor spent nuclear fuel. We have received input on technical strategies for the treatment, packaging, and disposal of research reactor spent fuel, as well as other spent fuel, from several interested organizations. The Department is in the process of preparing the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement*, which considers the potential environmental impacts of alternative means of managing spent fuel at the Savannah River Site. We will consider this information, as well as the views of Citizens for Nuclear Technology Awareness, cost, and nonproliferation policy concerns, prior to making a decision on the best path forward.

One of the concerns you raised was that the Department should maximize the use of existing processing facilities. Reprocessing of highly enriched aluminum research reactor fuel would, if carried out, take place in the Savannah River Site H-Canyon. This facility is currently operating to dissolve certain spent nuclear fuel in keeping with Departmental commitments made in response to Defense Nuclear Facilities Safety Board (DNSFB) Recommendation 94-1. In accordance with these and other commitments, H-Canyon would operate until the end of 2005 to address potential environment, health and safety vulnerabilities associated with the current form or storage configuration of certain spent nuclear fuels and other nuclear materials. In addition, the Department is currently evaluating potential canyon feed materials for which alternative disposition pathways may not be adequate. Were the Department to choose to process these materials in the H-Canyon, this could extend H-Canyon operations for an additional three to four years. Hence, it is possible that the facility would not be available until 2009 to reprocess aluminum spent fuel that does not present health and safety vulnerabilities.

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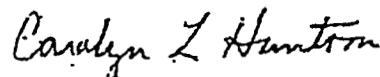
We believe that over the long-term (i.e., 20 to 30 years), annual spent fuel receipts at the Savannah River Site will be limited and thus it would not be cost-effective to continue to maintain canyon operations during the duration of these receipts. Therefore, the Department's preferred approach would be to construct a smaller more cost-effective facility for the treatment and management of aluminum spent fuel. We do plan, however, to ensure the continued availability of the Savannah River Site canyon processing facilities until we have successfully demonstrated implementation of an alternate treatment technology.

Concerning the safety of new technologies, the Department is in the process of preparing a response to Defense Nuclear Facilities Safety Board Technical Report 22, and will forward a copy of the response to you when it is completed. Should the Department decide to proceed with implementation of any alternative technology, we plan to put in place an independent review program to provide continuing expert assessment of the effectiveness and safety of any new technology.

Finally, with regard to the generation and subsequent disposal of waste from an alternative like melt and dilute, any meaningful comparison among treatment methods of the waste volumes that would need to be disposed of should include transuranic and low-level waste, not just high-level waste. As shown in the *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement*, both transuranic and low-level waste, as well as high-level waste, generated by reprocessing spent fuel in the canyons would be significantly greater than if the fuel were treated using the melt and dilute technology.

Please be assured that the Department will give fair consideration to reprocessing in our evaluation of how best to manage spent nuclear fuel at the Savannah River Site. If you have any questions, please contact Mr. David Huizenga of my staff at (202) 586-5151.

Sincerely,



Carolyn L. Huntoon
Assistant Secretary for
Environmental Management

Comment Letter L-19
State of South Carolina
State Budget and Control Board
February 11, 1999

STATE OF SOUTH CAROLINA
State Budget and Control Board
OFFICE OF STATE BUDGET

DAVID M. BEASLEY, CHAIRMAN
GOVERNOR

RICHARD A. ECKSTROM
STATE TREASURER

EARLE E. MORRIS, JR.
COMPTROLLER GENERAL

1122 LADY STREET, 12TH FLOOR
COLUMBIA, SOUTH CAROLINA 29201
(803) 734-2280

LES BOLES
DIRECTOR

JOHN DRUMMOND
CHAIRMAN, SENATE FINANCE COMMITTEE

HENRY E. BROWN, JR.
CHAIRMAN, WAYS AND MEANS COMMITTEE

LUTHER F. CARTER
EXECUTIVE DIRECTOR

February 11, 1999

Ms. Carol M. Borgstrom
Director, Office of NEPA Policy & Assistance
US Department of Energy (EIS-981215-015)
1000 Independence Avenue, S.W.
Washington, DC 20585

Project Name: Savannah River Site Spent Nuclear Fuel Management Draft environmental
Impact Statement select the appropriate treatment of packaging technology to prepare
aluminum-based spent nuclear fuel at SRS for ultimate disposition;

Project Number: EIS-981215-015

Dear Mr. Borgstrom,

The Office of State Budget, has conducted an intergovernmental review on the
above referenced activity as provided by Presidential Executive Order 12372. All
comments received as a result of the review are enclosed for your use.

The State Application Identifier number indicated above should be used in any future
correspondence with this office. If you have any questions call me at (803) 734-0494.

Sincerely,


Omega Burgess
Grants Services Coordinator

Fax: (803) 734-0645



Office of State Budget
South Carolina Project Notification and Review System
1122 Lady Street, 12th floor
Columbia, SC 29201

State Application Identifier
EIS-981216-016

Suspense Date
1/31/99

Stan M. McKinney
Office of the Adjutant General

RECEIVED
JAN 05 1998
Emergency Preparedness Division
Office of the Adjutant General

The Office of State Budget is authorized to operate the South Carolina Project Notification and Review System (SCPNRS). Through the system the appropriate state and local officials are given the opportunity to review, comment, and be involved in efforts to obtain and use federal assistance, and to assess the relationship of proposals to their plans and programs.

Please review the attached information, mindful of the impact it may have on your agency's goals and objectives. Document the results of your review in the space provided. Return your response to us by the suspense date indicated above. Your comments will be reviewed and utilized in making the official state recommendation concerning the project. The recommendation will be forwarded to the cognizant federal agency.

Should you have no comment, please return the form signed and dated.

If you have any questions, call me at (803) 734-0488

RECEIVED

☒

Project is consistent with our goals and objectives.

☐

Request a conference to discuss comments.

☐

Please discontinue sending projects with this CFDA# to our office for review.

☐

Comments on proposed Application are as follows:

Signature: _____

Stan M. McKinney

Date: 01/29/99

Title: Director, Emergency Prep. Division

Phone: (803) 734-8020



Office of State Budget

South Carolina Project Notification and Review System

1122 Lady Street, 12th floor
Columbia, SC 29201

State Application Identifier
EIS-981216-016

Suspense Date
1/31/99

Cornelia Gibbions
Governor's Division of Health & Human Services

The Office of State Budget is authorized to operate the South Carolina Project Notification and Review System (SCPQRS). Through the system the appropriate state and local officials are given the opportunity to review, comment, and be involved in efforts to obtain and use federal assistance, and to assess the relationship of proposals to their plans and programs.

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Should you have no comment, please return the form signed and dated.

If you have any questions, call me at (803) 734-0485.

☒

Project is consistent with our goals and objectives.

☐

Request a conference to discuss comments.

☐

Please discontinue sending projects with this CFDA# to our office for review.

☐

Comments on proposed Application are as follows:

Signature: Cornelia Gibbions

Date: 1/11/99

Title: _____

Phone: _____

RECEIVED
JAN 12 1999
Rodney C. Brown
Budget Control Board
OFFICE OF STATE BUDGET



Office of State Budget
South Carolina Project Notification and Review System

1122 Lady Street, 12th floor
Columbia, SC 29201

State Application Identifier
EIS-981216-016

Suspense Date
1/31/99

Robert Thomas
S. C. State Housing Finance & Development Authority

The Office of State Budget is authorized to operate the South Carolina Project Notification and Review System (SCPNRS). Through the system the appropriate state and local officials are given the opportunity to review, comment, and be involved in efforts to obtain and use federal assistance, and to assess the relationship of proposals to their plans and programs.

Please review the attached information, mindful of the impact it may have on your agency's goals and objectives. Document the results of your review in the space provided. Return your response to us by the suspense date indicated above. Your comments will be reviewed and utilized in making the official state recommendation concerning the project. The recommendation will be forwarded to the cognizant federal agency.

Should you have no comment, please return the form signed and dated.

If you have any questions, call me at (803) 734-0485.

☒

Project is consistent with our goals and objectives.

☐

Request a conference to discuss comments.

☐

Please discontinue sending projects with this CFDA# to our office for review.

☐

Comments on proposed Application are as follows:

Signature:

Robert Thomas

Date:

1/12/99

Title:

Director, Housing Development

Phone:

734-2150

RECEIVED
JAN 12 1999
Rodney Grizzle
Budget & Control Board
OFFICE OF STATE BUDGET



Office of State Budget

South Carolina Project Notification and Review System

1122 Lady Street, 12th floor
Columbia, SC 29201

State Application Identifier
EIS-981216-016

Suspense Date
1/31/99

Joel T. Cassidy
South Carolina Employment Security Commission

The Office of State Budget is authorized to operate the South Carolina Project Notification and Review System (SCPNRS). Through the system the appropriate state and local officials are given the opportunity to review, comment, and be involved in efforts to obtain and use federal assistance, and to assess the relationship of proposals to their plans and programs.

Please review the attached information, mindful of the impact it may have on your agency's goals and objectives. Document the results of your review in the space provided. Return your response to us by the suspense date indicated above. Your comments will be reviewed and utilized in making the official state recommendation concerning the project. The recommendation will be forwarded to the cognizant federal agency.

Should you have no comment, please return the form signed and dated.

If you have any questions, call me at (803) 734-0485.

Rodney Grizzle



Project is consistent with our goals and objectives.



Request a conference to discuss comments.



Please discontinue sending projects with this CFDA# to our office for review.



Comments on proposed Application are as follows:

Signature: _____

Joel T. Cassidy

Date: January 6, 1999

Title: _____

Executive Director

Phone: 803-737-2617



Office of State Budget
South Carolina Project Notification and Review System

1122 Lady Street, 12th floor
Columbia, SC 29201

State Application Identifier
EIS-981216-016

Suspense Date
1/31/99

George Bistany
South Carolina Department of Commerce

The Office of State Budget is authorized to operate the South Carolina Project Notification and Review System (SCPNSR). Through the system the appropriate state and local officials are given the opportunity to review, comment, and be involved in efforts to obtain and use federal assistance, and to assess the relationship of proposals to their plans and programs.

Please review the attached information, mindful of the impact it may have on your agency's goals and objectives. Document the results of your review in the space provided. Return your response to us by the suspense date indicated above. Your comments will be reviewed and utilized in making the official state recommendation concerning the project. The recommendation will be forwarded to the cognizant federal agency.

Should you have no comment, please return the form signed and dated **RECEIVED**

If you have any questions, call me at (803) 734-0485. **JAN 19 1999** Rodney C. Zelle

- ☐ Project is consistent with our goals and objectives. **Budget & Control Board**
OFFICE OF STATE BUDGET
- ☐ Request a conference to discuss comments.
- ☐ Please discontinue sending projects with this CFDA# to our office for review.
- ☐ Comments on proposed Application are as follows:

Signature: George Bistany

Date: 1-14-99

Title: Grant Manager

Phone: 734-0651

JAN 06 1999

DOE Response to Comment Letter L-18

Comments Noted

Comment Letter L-20
South Carolian Department of Natural Resources
January 28, 1999

South Carolina Department of Natural Resources



Paul A. Sandifer Ph.D.
Director

January 28, 1999

Mr. Karl Waltzer
Department of Energy
Savannah River Operations Office
Post Office Box A
Aiken, South Carolina 29802

Subject: Spent Nuclear Fuel Management
DOE/EIS-0279D

Dear Mr. Grainger:

The South Carolina Department of Natural Resources has received the December 1998 reports entitled *Nonproliferation Impacts Assessment for the Management of The Savannah River Site Aluminum-Based Spent Nuclear Fuel* and *Report on the Savannah River Site Aluminum-Based Spent Nuclear Fuel Alternatives Cost Study*. The Department does not have staff with the appropriate technical expertise to conduct a thorough analysis of these reports.

Sincerely,

A handwritten signature in dark ink, appearing to read "Robert E. Duncan", is written over a horizontal line.

cc Robert E. Duncan
Environmental Programs Director

DOE Response to Comment Letter L-20

Comments Noted

Comment Letter L-21

National Oceanic and Atmospheric Administration

June 22, 1999



UNITED STATES DEPARTMENT OF COMMERCE
National Oceanic and Atmospheric Administration
NATIONAL MARINE FISHERIES SERVICE

Southeast Regional Office
9721 Executive Center Drive North
St. Petersburg, FL 33702
(727) 570-5312; FAX 570-5517

JUN 22 1999

F/SER3:BH

Mr. Andrew Grainger
Department of Energy
Savannah River Operations Office
P.O. Box A
Aiken, South Carolina 29802

Dear Mr. Grainger:

This is in reference to your December 21, 1998 letter transmitting the Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement (DEIS). The DEIS analyzes various alternatives for the safe and efficient management of spent nuclear fuel and targets stored and scheduled to be received at the Savannah River Site (SRS), South Carolina, including the placement of these materials in a form suitable for disposition. The Department of Energy (DOE) has identified the melt and dilute option as the preferred alternative for managing most of the aluminum based spent nuclear fuel with conventional processing as the preferred method for 3% of the spent fuels.

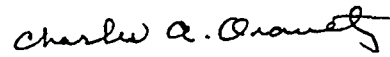
The action area for the proposed action and alternatives is adjacent to the Savannah River which has one of 20 distinct population segments of the shortnose sturgeon (*Acipenser brevirostrum*). The shortnose sturgeon is listed as endangered by the Endangered Species Act (ESA). According to the DEIS the proposed action and alternatives would utilize existing upland areas and would not affect the Savannah River or its tributaries. The DEIS also states that DOE will comply with all Federal and state requirements for the containment and discharge of surface water and wastes from the action area. Based on these factors, the National Marine Fisheries Service (NMFS) does not believe that the proposed action and alternatives would affect the shortnose sturgeon nor any other species protected by the ESA under NMFS purview.

This concludes the DOE's consultation responsibilities under section 7 of the ESA for the proposed action and alternatives described in the December 1998 DEIS for species under NMFS purview. Consultation should be reinitiated if new information reveals impacts of the identified activity that may affect listed species or their critical habitat, a new species is listed, the identified activity is subsequently modified or critical habitat determined that may be affected by the proposed activity.



If you have any questions, please contact Bob Hoffman, Fishery Biologist at (727) 570-5312.

Sincerely yours,



Charles A. Oravetz
Assistant Regional Administrator
Protected Resources Division

cc: F/PR3

Response to Comment Letter L-21

Comments noted; See also response to Comment L11-2

DOE Public Meetings on Draft EIS

Columbia, SC: January 28, 1999

North Augusta, SC: February 2, 1999

PUBLIC MEETINGS

The public meetings consisted primarily of informal discussions and questions and answers related to Spent Nuclear Fuel (SNF) Management. In this section, each public meeting speaker's statement is placed in context and paraphrased because some statements are dependent on previous statements and interspersed with other discussion. The transcripts from the meetings can be reviewed at the DOE public reading rooms: DOE Freedom of Information Reading Room, Forrestal Building, Room 1E-190, 1000 Independence Ave., S.W., Washington, D.C., 20585, Phone: 202-586-6020 and DOE Public Document Room, University of South Carolina, Aiken Campus, University Library, 2nd Floor, 171 University Parkway, Aiken, S.C. 29801, Phone: 803-648-6851.

M1-01: During DOE's presentation of SNF packaging technologies, one commenter requested that DOE clarify the difference in direct disposal and direct codisposal in the *Prepare for Direct Disposal/Direct Co-Disposal* packaging technology option.

Response: Section 2.2.3.1 presents this option. Under direct co-disposal, DOE would co-dispose canisters of vitrified high-level waste and a canister of SNF in a repository waste package. Under direct disposal, DOE would dispose of the fuel between waste packages of commercial SNF.

These actions would take place at an offsite repository. From an SRS perspective, Direct Co-disposal is identical to Direct Disposal, except for possible differences in the diameter of the canisters into which the SNF would be placed. In either case, the canisters would be shipped to the repository in shipping casks and repackaged into repository packages.

M1-02: One commenter asked why DOE expected receipts of domestic reactor fuel to decrease to the point that operating the canyons (i.e., uranium and plutonium separations facilities) would be inefficient.

Response: The relatively large domestic research reactor fuel receipt rates before the year 2009 are primarily due to shipment of the inventory of SNF that has already been received by DOE and is being stored at DOE sites. After the year 2009, DOE would have shipped its entire existing domestic SNF inventory to SRS and would then only ship SNF as it is generated (i.e., removed from use in research reactors). DOE anticipates that the receipt rate after 2009 would consist of approximately 150 Materials Test Reactor-like elements and 12 High Flux Isotope Reactor assemblies per year.

M1-03: One commenter read part of his prepared comments regarding the continued operation of the canyons. He stated that it costs approximately \$400 million per year to maintain and operate the processing canyons at SRS, they were designed to operate at high levels, it is a waste of money to operate them for research reactor fuel, and it is the technology that generates the most radioactive waste.

M1-04: The commenter continued and stated that operating the canyons (a Cold War technology) would set a poor example to the world that the U.S. is serious about stopping the spread of weapons of mass destruction.

M1-05: The commenter expressed his concern that if the canyons operate for a small amount of fuel, DOE will consider proposals to bring more waste for processing (i.e., it is a temptation for DOE to find other missions for the canyons).

Response to comments M1-03, -04, and -05: As stated in Section 2.4.3 and shown in the accompanying figure entitled *The Preferred Alternative Management Flow Path* it is DOE's intent to complete processing in the canyons in the next few years. It is not DOE's intent to operate the canyons for processing SNF that is not already at SRS or expected to be received at SRS (see Section 2.1).

M1-06: One commenter indicated that she was interested in incidence of cancer and asked if the male workers were more at risk for developing

certain types of cancers (e.g., prostate or leukemia).

Response: DOE calculates adverse health effects to workers and the general public in terms of an "effective dose equivalent," which is organ doses weighted to yield equivalent whole-body doses. The effective dose equivalent is used to calculate an estimated number of total fatal cancers, which does not differentiate between specific types of cancers. The effective dose equivalent also does not differentiate between genders.

M1-07: One commenter asked about the dose reconstruction studies of the Centers for Disease Control and Prevention (CDC), and if this information would affect evaluations in this EIS.

M1-08: After discussion on the CDC study and the release data being used, the commenter stated that it is not the release data that is his concern. It is the effect of those releases.

Response: The CDC is performing a dose reconstruction study at SRS. Phase I of the study developed releases numbers. Phase II of the study will look at dose affects. Copies of the report can be obtained from the CDC (Paul Renard, 4770 Buford Highway NE, M.S. F-35, Atlanta, GA, 30341 or by phone at 770-488-7030).

The recent draft report by the Centers for Disease Control and Prevention (CDC) concludes that emissions of certain radionuclides from the Savannah River Site were underreported, primarily in the 1950s and early 1960s. The CDC report states that its estimates for the time since the late 1960s are in close agreement with DOE estimates made at the time. Neither the CDC nor DOE has data on the cumulative health impacts.

In this EIS, DOE has predicted the effect of potential radioactive material releases from normal operations and potential accidents for all the SNF treatment options considered for managing SNF

at SRS. For normal operations, the release estimates were based on recent site emissions data that are publicly reported on an annual basis. This information is discussed in Section 3.7 of the EIS. The results of the CDC study do not affect any of the determinations made in this EIS because projections of radioactive material releases are derived from recent emissions data that the CDC report concludes are in close agreement with its estimates.

M1-09: One commenter asked about the possibility of explosions due to accidental production of hydrogen.

Response: Potential explosions, such as hydrogen-initiated explosions, were included in the accident analysis (see Appendix D). This accident analysis is based on safety analyses and historical facility data. For new facilities without design details, DOE based the accident analysis on hazard analyses, historical data for similar facilities and operations, and best estimates.

Hydrogen explosions require the generation of hydrogen (from the radiolytic decomposition of water or hydrogen-rich materials), the failure of ventilation systems designed to prevent build-up of hydrogen, and an ignition source. The combination of these events makes a hydrogen deflagration an extremely unlikely event (once in 18,000 years). Refer to the Table in Appendix D on H-Canyon radiological accidents and impacts.

M1-10: The commenter asked for the quantity of plutonium that would be produced from these chemical processing activities under the Preferred Alternative.

Response: The amount of plutonium that would be recovered from the Experimental Breeder Reactor-II fuel (the major source of plutonium among SNF types assessed in the EIS) and the Mark-42 targets under the preferred alternative is approximately 114 kilograms. No other plutonium would be recovered.

TC

Any plutonium recovered from SNF by the Conventional Processing technology would be considered surplus to the nation's nuclear weapons program and would be placed in storage at the SRS pending disposition under decisions DOE will reach as part of the Surplus Plutonium Disposition Environmental Impact Statement currently being prepared.

M1-11: While a DOE representative was discussing DOE's preference and schedule for the melt and dilute process, a commenter asked if there would be sufficient funds to develop and implement this technology.

Response: DOE has conducted laboratory testing for the melt and dilute process. Design completion of a pilot scale facility (L-Area Experimental Facility LEF) is scheduled by the end of FY 2000. The design and construction of a full-scale facility would need to be developed in the context of constrained, out-year budget targets, and funding for such a facility would need to be balanced against other priorities at SRS. DOE has developed a schedule that can be used as a baseline for near-term planning and budgeting purposes. LEF is scheduled to be constructed and online by the end of FY 2002.

M1-12: One commenter asked if there was a time limit on the canyons' life span and if that was one reason for the preference to proceed with the development of the melt and dilute technology.

Response: The SRS canyons are currently operating to stabilize nuclear material that is outside the scope of this EIS. DOE has not identified the useful life of the canyons, however, the current canyon planning basis does establish an endpoint for operation activities. The canyons are expected to cease operations before the future receipts of additional spent nuclear fuel are completed.

M1-13: One commenter read text prepared by another commenter that included this comment and comment M1-14. The commenter stated that one drawback to the melt and dilute technology is that SRS has repeatedly refused to

request full funding to develop the technology, leaving DOE officials in Washington, D.C. to add it to the SRS budget. He stated that this reflects internal DOE politics and the preference of many to continue processing in the canyons as long as possible.

Response: DOE has conducted laboratory testing for the melt and dilute process and plans to begin a pilot scale facility test program next fiscal year. The design and construction of a full-scale facility would need to be developed in the context of constrained, out-year budget targets, and funding for such a facility would need to be balanced against other priorities at SRS. DOE has developed a schedule that can be used as a baseline for near-term planning and budgeting purposes. The FY 2000 funding for SRS has been established, and includes funding for the design completion of the pilot scale facility. The pilot scale facility is scheduled to be constructed and placed online by end of FY 2002.

M1-14: This commenter also stated that a second drawback dealt with controlling volatilized radioactive material and the technology cannot be safely implemented until DOE proves that it can recover most of the radionuclides and prevent their release to the environment.

Response: DOE would not implement any SNF treatment technology until the safe operation of the technology has been established through testing and engineering design. During fiscal year 2000 a demonstration of the off-gas system (which is designed to capture volatilized radionuclides) is expected to resolve most technical uncertainties.

M1-15: One commenter asked why uranium recovered by conventional processing of Group A fuels would be made available for commercial use while the uranium in Groups B, C, and D would be immobilized and left in the waste form.

Response: DOE's rationale for applying Conventional Processing is to stabilize fuels that are at risk, not to recover the uranium in the fuel. However, due to the design of the SRS canyons,

recovery of uranium is an inescapable part of the process. Recovered low-enriched uranium has potential commercial value and could be made available for sale. Group B and the vast majority of Groups C and D do not present potential health and safety vulnerabilities and would be treated in the Preferred Alternative by a non-chemical separations process technology (Melt and Dilute) that does not separate the uranium from the fuel.

M1-16: One commenter asked if any of the yet-to-be received SNF would be processed in the SRS canyons.

Response: In this EIS, DOE has not proposed to process any yet-to-be-received SNF in the SRS canyons other than powdered/oxide fuel that may be received at SRS while H canyon is still in operation. Specifics of the material that would be treated by conventional processing are in Section 2.4.3.2 of the EIS.

M1-17: One commenter asked if DOE has considered dry cask storage for Fuel Groups E (higher actinide targets) and F (non-aluminum-clad fuels to be shipped to INEEL).

Response: DOE considered dry storage as a component of the alternative, Repackage and Prepare to Ship. Dry storage was considered as a potential intermediate step in preparing Mark-51's and other targets from Fuel Group E for offsite shipment (for programmatic use). Dry-storage is still considered to be a viable, intermediate step for Fuel Group F if it is available prior to Fuel Group F being shipped offsite. Fuel Group F will remain in wet-storage until a dry-storage facility is available and/or until it is shipped offsite. For other fuel groups that will be treated at SRS dry storage is not considered a reasonable alternative because it would not prepare the spent nuclear fuel for disposal in a repository. See Section 2.4.6 in the EIS.

M1-18: One commenter submitted four recommendations as paraphrased in comments M1-18 through M1-21. The first recommendation had four components. The commenter stated that (1) DOE should accelerate funding of the melt

and dilute technology, (2) determine whether volatilized radionuclides can be captured in the facility's off-gas system, (3) make all information publicly available, and (4) initiate an independent review panel to evaluate engineering design.

Response: DOE has plans in place to be responsive to the four components of this suggestion. As indicated in the response to comment M1-11, under existing funds, DOE has conducted laboratory testing of the technology. The design and construction of a full-scale facility would need to be developed in the context of constrained, out-year budget targets, and funding for such a facility would need to be balanced against other priorities at SRS. DOE has developed a schedule that can be used as a baseline for near-term planning and budgeting purposes. The FY 2000 budget has been established and includes funding for the design completion of the L-Area Experimental Facility (LEF). LEF will demonstrate feasibility of the melt and dilute technology. LEF is scheduled to be online by the end of FY 2002.

As detailed in the response to comment L2-17, development of an off-gas system is part of the ongoing laboratory experiments and testing, and further testing will be accomplished in the pilot project phase of the melt and dilute system development program. As detailed in the response to comment L2-20, DOE-SR encouraged full and open public review through the scoping period which preceded preparation of the Draft EIS and the public comment period on this Draft EIS. Finally, DOE is currently planning for independent review of the proposed melt and dilute project.

M1-19: In the second recommendation, the commenter stated that based on its decision to phase out processing, DOE should only process to resolve clearly demonstrated and imminent hazards and that DOE should take the lead in demonstrating that nuclear materials can be safely managed without separating weapons-usable material.

Response: DOE fully supports its decision to phase out processing and to develop and safely operate technologies that prepare SNF for permanent disposal using a non-chemical separations based process. However, DOE does not believe it is prudent to wait for an imminent hazard before chemically processing some SNF. Instead, DOE proposes to process a relatively small volume (about 3 percent by volume and 40 percent by mass) of aluminum-based SNF. The rationale for this processing is to avoid the possibility of future urgent actions, including expensive recovery actions that would entail unnecessary radiation exposure to workers, and in one case, to manage a unique waste form (i.e., core filter block).

M1-20: The third recommendation was to include in the Final EIS a clear comparison of potential future processing rates, costs, and waste generation rates with historic levels. The comparison should allow independent comparison of processing rates and whether the SRS processing canyons are appropriate for the proposed activities.

Response: DOE used actual historic SRS data, for canyon operations (e.g., processing rates [McWhorter 1997], waste generation rates [Bickford, et. al. 1997], and costs [DOE 1997]) when preparing this EIS. DOE strives to make all data and information available for members of the public and other interested parties. All reports and technical documents used to support the analysis in the EIS are available for review at the DOE public reading rooms at the Gregg-Graniteville Library of the University of South Carolina-Aiken, in Aiken, South Carolina, and in Room 1E-190 at DOE Headquarters in the Forrestal Building, 1000 Independence Ave, Washington, DC.

M1-21: The fourth recommendation was to relate the actions proposed in the EIS with all other pending decisions regarding future processing activities, decommissioning of the processing facilities, and future actions for managing nuclear materials at SRS.

Response: As described in the response to comment L2-16, pending actions and decisions that could impact SRS SNF decisions (e.g., Rocky Flats Plutonium Residues and Scrub Alloy EIS, Surplus Plutonium Disposition EIS) are considered in the cumulative impacts presented in Chapter 5 of the EIS and in DOE's planning efforts for SRS.

The Process Needs Assessment (PNA; see Section 1.6.2 of the EIS), recommended that the SRS canyons be considered for several materials at various DOE sites. Among those were the Mark-18 and Mark-42 targets at the SRS. These targets have been included in this EIS. For the majority of the other materials, the canyons were identified primarily as a backup alternative.

DOE is continuing to evaluate the disposition pathways for these materials to determine which, if any, should be proposed for canyon processing. Prior to making any decisions for the further use of the canyons to support the Department's needs, analysis performed in accordance with the National Environmental Policy Act will be completed. The PNA is a planning document from which proposals for further processing may develop.

Other material discussed for processing at SRS under the PNA include single-pass reactor SNF at Hanford, and a small amount of damaged SNF at INEEL, classified fissile material metal parts at RFETS, and plutonium scrap at Hanford. Currently, DOE has no plan or proposal to transfer the single-pass reactor SNF at Hanford or the damaged SNF at INEEL to SRS so that material was not considered for the cumulative impacts under this EIS. In an amended Record of Decision for the *Final Environmental Impact Statement on Storage and Disposition of Surplus Fissile Material*, DOE decided to transfer classified metal from RFETS to SRS for stabilization and storage. DOE is considering transferring the plutonium scrap from Hanford to SRS for stabilization and storage pending appropriate National Environmental Policy Act review. As a result, DOE has included proc-

essing that material as part of the cumulative impacts for this EIS.

DOE is continuing to evaluate the inventory of nuclear material at facilities throughout the DDE complex. DOE's Nuclear Material Integration initiative is one such recent effort that has identified material which could be processed at the SRS. Although there are no current plans to process these materials at SRS, DOE considers it appropriate to include a qualitative estimate of impacts as part of the cumulative impacts for this EIS because it is not unforeseen that processing at SRS could occur.

DOE has also included, processing about 56 MTHM of de-clad sodium-bonded fuel from INEEL as part of the cumulative impacts analysis in this EIS. Because the *Treatment and Management of Sodium-Bonded Spent Nuclear Fuel EIS* (now being developed) describes processing that material at SRS as a reasonable alternative.

M1-22: One commenter inquired about the status of the management of Fuel Group F at INEEL.

Response: DOE plans to begin shipments of non-aluminum-based SNF from SRS to INEEL in 2008 and to complete the shipments in 2013. INEEL included the NEPA analysis for the stainless steel/zirconium-clad fuel that makes up Fuel Group F in the *Final Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management EIS* (DOE 1995).

M1-23 through M1-32: One commenter provided a six-page statement for the record. DOE has responded to comment M1-23, 24, 25, and 27 as a group and provided separate responses for the others. His comments are paraphrased as follows:

M1-23: The Draft EIS does not provide an assured path for disposition of aluminum-based SNF at the national repository.

M1-24: There are two deficiencies. First, waste acceptance criteria for placement of spent nuclear fuel into the geologic repository is not available.

M1-25: Secondly, the Draft EIS does not assess the performance of each of the alternatives against the current waste acceptance system requirements or against potential waste acceptance criteria. The draft assumes that each technology can prepare a road-ready package that is acceptable at the repository; it lacks information on the conformity of technology-specific waste forms with package requirements for the repository.

M1-26: Only conventional processing has the capability for final disposition and the draft does not state when final waste acceptance criteria for SNF will be available nor when the acceptability of the waste forms from the alternative technology options will be ascertained.

Response: DOE is confident that the preferred alternative described in the EIS will result in the ability to dispose of aluminum-based spent nuclear fuels in any potential geologic repository. Preliminary repository waste acceptance criteria have been established by DOE's Office of Civilian Radioactive Waste Management. These criteria form the basis for the technology comparisons made in this EIS. While preliminary, the repository requirements are well understood by DOE and will be finalized during the Nuclear Regulatory Commission (NRC) review of the repository operating license application. The results of this review are expected in late 2009.

M1-27: There is no discussion or assurance that a treatment technology option will meet the repository acceptance criteria before conventional processing is shut down.

Response to comments M1-23, 24, 25, and 27: DOE-SR is working closely with the Nuclear Regulatory Commission (the federal agency that would license the operation of a geologic repository) to ensure that the final product from the selected SNF treatment technology would be acceptable for disposition. The EIS has been

revised to discuss in greater detail the expected repository acceptance criteria and compare the treatment technology products to those criteria. This information is discussed in Section 2.2.1.

Recognizing that repository disposal is the ultimate endpoint for the melt and dilute waste form, DOE-SR signed in August 1997 a Memorandum of Understanding with NRC for their review and feedback on the research effort that DOE-SR is conducting. DOE-SR has provided the NRC several technical reports on the results obtained from the research effort. Based upon their initial review, the NRC in a June 1998 letter stated that "both the direct co-disposal and melt-dilute options would be acceptable concepts for the disposal of aluminum-based research reactor SNF in the repository." Additionally, as research efforts yield new findings, DOE-SR is providing the information to the NRC for their feedback and review.

M1-28: A significant health, safety, and environmental risk will eventually result if none of the technology options can meet the repository acceptance criteria and the canyons are permanently shut down.

Response: Health, safety, and environmental risks associated with continued storage were estimated for the No-Action Alternative. Also note that DOE has scheduled to bring online a pilot scale facility, L-Area Experimental Facility (LEF), in 2002 to demonstrate the melt and dilute technology.

M1-29: The EIS should be modified to include final SNF acceptance criteria and an evaluation of the technology options against these criteria. The preferred technology(s) should be limited to those that have a certainty of acceptance similar to the DWPF vitrified waste.

Response: The comments are beyond the scope of the EIS. Parallel, but separate, NEPA documentation is ongoing for the geologic repository.

M1-30: DOE should maintain the operability of the canyons until a technology option has been selected and demonstrated.

Response: Under the current planning basis DOE will have high confidence in the acceptability of the waste form before the canyons cease operations.

M1-31: If DOE cannot provide assurance of maintaining canyon operability until a technology option has been selected and demonstrated, the EIS should include the safety, health and environmental impacts of long-term interim storage and required maintenance to ensure integrity of the waste forms.

Response: The situation described in the comment was evaluated under the No-Action alternative.

M1-32: The draft EIS and its supporting cost study should include the value of uranium credits.

Response: The cost study referenced in the EIS states that uranium credits in the range of \$110 million to \$150 million were considered in developing the cost report. However, due to the recently signed agreement between Russia and the United States that calls for the potential deferment of HEU sales, no actual value was assigned in the report for calculating life-cycle costs associated with processing activities. The EIS has been revised to clarify this point (see Section 2.6.5).

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M1-33 through M1-42: Another commenter also provided a six-page statement for the record. DOE has responded to some comments as a group and has provided separate responses for the others. His comments are paraphrased as follows:

M1-33: Avoid any technology that involves further processing of SNF. Address the apparently temporary problem of what DOE characterizes as failed fuels that pose imminent health risks.

M1-34: This EIS process has failed to acknowledge the strong institutional bias for the continued operation of the canyons.

Response: As detailed in the response to comment M1-19, DOE's proposed action is to select a new non-chemical processing technology that would put aluminum-based SNF into a form or container suitable for direct placement in a geologic repository. DOE would apply the Conventional Processing technology only to a relatively small volume (about 3 percent by volume and 40 percent by mass) of SNF at SRS. The rationale for this processing is to avoid the possibility of future urgent actions, including expensive recovery actions that would entail unnecessary radiation exposure to workers, and in one case, to manage a unique waste form (i.e., core filter block).

M1-35: There should be ways to bridge the period of time between now and when an alternative treatment technology is available that do not require operation of the canyons.

Response: DOE analyzed not treating the SNF during this period with the precaution of repackaging it in the no action alternative.

M1-36: DOE has not determined the long-term occupational and public health effects of operating the canyons but DOE continues its bias to keep them operating.

Response: DOE has not identified any long-term (>50 years) occupational or public health effects. The Centers for Disease Control (CDC) is performing a dose reconstruction study at SRS. Phase I of the study developed releases numbers. Phase II of the study will look at dose affects. Copies of the report can be obtained from the CDC (Paul Renard, 4770 Buford Highway NE, M.S. F-35, Atlanta, GA, 30341 or by phone at 770-488-7030).

The occupational and public health impacts presented in the EIS for both normal operations and potential accidents represent projections of future SNF management activities. Because of the inherent conservatism built into the analysis,

DOE believes that the actual impacts would be less than those reported in the EIS. DOE relied on data from existing facility operations as much as possible in performing the analyses presented in the EIS. For example, for normal operations, the release estimates were based on recent site emissions data that is publicly reported on an annual basis.

Annual emission data are provided in the SRS Environmental Report which is available in the DOE public reading room in the Gregg-Graniteville Library at the University of South Carolina – Aiken, Aiken, S.C., or on the web at <http://www.srs.gov/general/srenviro/endrpt/index.html>.

M1-37: The melt and dilute proposal, like chemical processing, has a pathway to release radioactive materials to the environment and concepts for control of these releases remain unproven.

Response: DOE's objective is to develop an off-gas system that will ensure the safety of workers, the public, and the environment; comply with DOE Orders and Standards; and meet all regulatory requirements. Emissions from any one operating facility are normally a fraction of the applicable Site limits. Development of an off-gas system is currently in progress based on results from laboratory experiments and testing. Further testing will be accomplished during the pilot project phase of the melt and dilute system development program that is scheduled to be conducted by the end of FY 2002.

M1-38: Pursue dry cask passive cooled storage as a way to bridge the time between the present and the availability of long-term repository storage.

Response: DOE's objective for management of aluminum-based SNF at SRS is two-fold: to provide safe and efficient interim SNF storage and to prepare the SNF for placement in a geologic repository. Transferring the aluminum-based SNF to dry-storage without treatment would not satisfy DOE's objectives. The dry-

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storage alternative was evaluated in the EIS but dismissed as discussed in Section 2.4.6. Note that dry storage after SNF treatment is complete is a component of all new SNF treatment technologies that DOE considered in this EIS.

M1-39: In accordance with the president's executive order on environmental justice, DOE must prepare an analysis to determine whether minority or low-income communities could receive disproportionately high and adverse impacts as a result of actions proposed in the EIS. It is not apparent that this has been done in this environmental analysis.

Response: DOE completed an environmental justice evaluation of the alternatives considered in this EIS and found there are no environmental justice concerns associated with the proposed action. The information is provided in Section 4.1.1.6 of the EIS.

M1-40: Dry cask storage seems inevitably to be preferable to continuing to operate the canyons and applying the melt and dilute technology, which could send a further stream of plutonium contaminated clothing to an industrial and nuclear laundry in Columbia, South Carolina. In 1995 the Interstate Nuclear Services (INS) began receiving plutonium-contaminated protective gear generated in the course of Savannah River Site's environmental restoration. DOE should not spread waste beyond the confines of its reservation.

Response: DOE assessed the impacts of an off-site laundry facility in a 1994 Environmental Assessment. DOE concluded, based on the known impacts of operation of the SRS laundry facility, that no latent cancer fatalities and no adverse health effects from chemical exposures would result from the operation of a licensed facility offsite. DOE issued a Finding of No Significant Impact (FONSI) in December 1994.

Current guidance on Environmental Justice and NEPA from the Council on Environmental Quality indicates that agencies are to consider whether effects are significant (as that term is defined under NEPA) when determining if im-

pacts are disproportionately high and adverse. DOE concluded that because health effects from the operation of a licensed laundry facility would not be significant, they also would not be disproportionately high and adverse to any segment of the population. While each of the alternatives addressed in the Spent Nuclear Fuel EIS (including No Action) would generate laundry that would be sent to an appropriately licensed and permitted facility, the small increase in volume would not affect the projected impacts.

In March, 1999, the court upheld Interstate Nuclear Service's operating permit.

M1-41: DOE should construct a nuclear laundry on SRS to replace the aging facility that it shut down.

Response: Comment noted.

M1-42: Halt the shipment of plutonium contaminated wastes to Columbia and do not increase the levels of those shipments by the choice of the current preferred alternative for SNF management.

Response: See response to comment M1-40.

M1-43: One commenter asked if DOE had plans for the year 2000 (Y2K) phenomenon.

Response: DOE's Y2K program identified vulnerable systems at SRS and made them Y2K compliant. No problems were identified during the roll over.

M1-44: One commenter stated that he was troubled over DOE continuing to advocate operation of the canyons.

Response: DOE is currently operating the canyons at SRS to stabilize nuclear material that has potential health and safety vulnerabilities. DOE proposes to use the canyons to process a small amount of SNF considered in this EIS for the same reason. The bases for this proposal are discussed in Section 2.4.3.2 of the EIS.

M1-45: One commenter asked if the cost study on the SNF alternatives included repository costs.

Response: Yes, the cost study included repository costs.

M2-01: One commenter asked if the melt and dilute technology produced chemical byproducts like conventional processing.

Response: The melt and dilute process would not result in the generation of chemical byproducts such as high-level waste. The end product of the melt and dilute process would be approximately 400 road-ready SNF canisters. The other secondary wastes generated by the melt and dilute process (e.g., low-level waste, transuranic waste) would be comparable to waste SRS currently handles and therefore, would not require unique treatment, storage, or disposal actions.

M2-02: The commenter asked if the goal of the melt and dilute technology was material for reuse or disposal of SNF in a geologic repository.

Response: See response to M2-01. The end product of melt and dilute is SNF canisters for disposal in a geologic repository.

M2-03: One commenter asked if DOE has any contingency plans in case the geologic repository is not built.

Response: DOE's current planning basis is to place SNF in a geologic repository. Secretary Richardson expressed his commitment to this plan during his July 22, 1998 confirmation testimony before the Senate Energy and Natural Resources Committee. DOE's proposed SNF treatment technologies would place SNF in a form suitable for safe interim storage at the SRS pending transfer to a geologic repository.

M2-04: DOE explained its intent to prepare the SNF in a form that is ready for shipment to a repository and the secondary benefit of achieving a stable waste form for interim storage. The commenter replied that this eliminated the no-

action alternative and questioned the need to stabilize the waste.

Response: It is DOE's policy to store this material in a form suitable for disposal so that, when a geologic repository becomes available, stored spent fuel could be shipped without delay.

M2-05: One commenter asked why DOE considers the melt and dilute to be the best technology option.

Response: As detailed in Section 2.4.3.1, DOE believes that the melt and dilute technology is the best choice among applicable technologies for the treatment of Material Test Reactor like fuel, most of the Loose Uranium-Oxide in Cans fuel, and most of the HEU/LEU Oxide and Silicide fuel. It satisfies DOE's objective and preference for a non-chemical separations-based technology and is fully compatible with and supportive of nonproliferation objectives. Melt and Dilute is preferred over the other non-separations based technologies because it is the most efficient technology in volume reduction; it has the flexibility to engineer the final waste form to provide a high-degree of confidence in its ability to be acceptable for placement in a geologic repository; it is relatively simple to implement; and it is less expensive than other similar technology options.

Another important discriminator is the technical maturity of the melt and dilute technology. DOE has conducted several years of research on the process, including actual melting of surrogate materials. A pilot test facility will be fabricated in the L-Reactor Area that will permit melting of irradiated spent nuclear fuel assemblies.

M2-06: One commenter asked if the product of melt and dilute (ingots) could be stolen and used in the production of nuclear weapons.

Response: The final melt and dilute product would be low-enriched uranium containing fission products and other materials such as aluminum. As a result, chemical separation and re-enrichment of the uranium would be required

before the materials could be used in weapons. Accordingly, in the nonproliferation report referenced in the EIS, DOE judged that the melt and dilute technology was fully consistent with nonproliferation policy.

M2-07: When DOE acknowledged that the melt and dilute requires a system to capture volatilized radioactivity in the offgas, one commenter asked if DOE had such a system.

Response: Development of an offgas system is currently in progress based on results from laboratory experiments and testing. Further testing will be accomplished during the pilot project phase of the melt and dilute system development program that is scheduled to be completed by the end of FY 2001.

M2-08: One commenter asked what DOE would do if it discovered that the off-gas system was not effective and, if DOE had to go to a backup option, would another EIS be required.

Response: If testing were to show that the proposed offgas system was not as effective as necessary, DOE would evaluate alternative offgas system designs.

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The current backup technology is the Direct co-disposal option. If DOE were to select this option or another SNF treatment technology evaluated in this EIS, it would not be required to prepare another EIS.

M2-09: One commenter asked if there were a technology, whether it be short-term or long-term, that could convert the SNF into a substance that did not require disposal in a geologic repository.

Response: There is no SNF management technology that would eliminate the high concentrations of radioactivity and associated high radiation levels in SNF. DOE's objective is to place the SNF in a form that is ready for disposal at a geologic repository. DOE believes that the melt-and-dilute technology achieves this objective.

M2-10: One commenter stated that Yucca Mountain (as the site for a geologic repository) did not make sense and stated that there was an earthquake there last week.

Response: Impacts from utilizing a geologic repository for disposal of commercial SNF and DOE SNF and high-level waste are currently being evaluated under the Yucca Mountain Repository EIS. These evaluations will include impacts from potential earthquakes.

M2-11: One commenter asked what amount and kind of waste would be generated by the melt and dilute process.

Response: As indicated in Section 4.1.1.4 of the EIS, melt and dilute would generate the following quantities of waste:

High-level waste – 2,200 cubic meters

Transuranic waste – 350 cubic meters

Mixed waste – 28 cubic meters

Low-level waste – 27,000 cubic meters

M2-12: One commenter asked if spent mixed-oxide fuel could be processed by the melt and dilute technology.

Response: DOE is analyzing management of aluminum-based SNF stored at SRS or expected to be shipped to SRS. Spent mixed-oxide fuel from commercial power reactors would be managed in the same manner as other commercial spent fuel, and would be sent directly to the repository for disposal. Commercial fuel would be disposed of directly in the repository because it is ceramic-based and zirconium-clad and so more durable than aluminum-clad fuel, and it is composed of low-enriched uranium that does not present a proliferation threat.

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M2-13: One commenter asked why would DOE sell LEU to a commercial producer of nuclear energy.

Response: The use of the Conventional Processing technology could result in the recovery of low-enriched uranium, which could be sold to a commercial producer of nuclear fuel. The alternative would be to dispose of the uranium as waste. This would be more costly and would create more secondary waste streams. However, DOE would not process any SNF for the express purpose of producing commercial nuclear power fuel feedstock. The purpose of processing SNF using chemical separations would be to address potential health and safety vulnerabilities.

M2-14: Continuing on her comment on uranium sales, the commenter stated that other countries (e.g., Germany) are moving away from nuclear power plants and the U.S. should not support nuclear power by providing LEU for use by commercial power companies.

Response: The purpose of the conventional processing technology is to quickly alleviate the potential health and safety concerns of that fuel. The only technology available to do that is the conventional processing capability of the canyons. The canyons produce a highly enriched uranium solution. This EIS proposes to take that a step further and bring it down to low-enriched uranium (LEU) so that it is not weapons-grade material any more. Options for dealing with the LEU are to continue to store it, convert it to an oxide and store it, or store it pending potential sale as feedstock for commercial nuclear fuel. The future of commercial nuclear power is not within the scope of this EIS.

M2-15: One commenter asked what environmental impact would result from the release of cesium into the atmosphere in the event that the filtration system doesn't capture all the cesium.

Response: In the EIS, DOE has analyzed a postulated accident that would result in a release of fission products, including cesium, from the melt and dilute process. This accident is calculated to result in an offsite collective radiation dose of 0.23 person-rem, which could result in much less than one latent cancer fatality (0.0001 latent cancer fatality).

M2-16: One commenter asked why DOE uses 50 miles as the radius for evaluating health effects.

Response: In the EIS, DOE evaluates health effects from airborne releases of radioactivity and chemicals to a distance of 50 miles in accordance with DOE Order 4500.5. If the analysis were to show any potential effects beyond 50 miles, then DOE would analyze further. Use of this constant interval also ensures comparability and consistency between SRS documents (i.e., EISs, Annual Environmental Reports).

For liquid releases of radioactivity and chemicals, DOE evaluates health effects to any population downstream from the release that could consume or otherwise come into contact with the receiving body of water. This population could reside at distances greater than 50 miles from the release point.

M2-17: One commenter noticed that three to five million gallons of (contaminated) water are in wet storage basins that would temporarily store SNF pending treatment. The commenter asked how DOE plans to treat this water or what DOE plans to do with this water.

Response: DOE would develop methods to safely disposition water in the storage basins as part of any program to decommission the storage basins. The system to remove water would be designed as part of the decontamination and decommissioning process.

M2-18 through M2-29. The DOE meeting facilitator asked if there were any further questions. Two commenters provided questions and statements paraphrased below.

M2-18: Which technology is considered processing?

Response: In the EIS, DOE is evaluating the Conventional Processing technology, which uses the SRS chemical separations facilities (F and H Canyons). When DOE was using these facilities for the production of nuclear materials

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for national defense, the process was referred to as "reprocessing."

M2-19: What percentage of the fuel would be processed?

Response: DOE proposes to apply the Conventional Processing technology to a relatively small volume of SNF at SRS (about 3 percent by volume and 40 percent by mass) before a new treatment facility is in place. The rationale for this processing is to avoid the possibility of future urgent actions, including expensive recovery actions that would entail unnecessary radiation exposure to workers, and in one case, to manage a unique waste form (i.e., core filter block).

M2-20: Why is DOE receiving fuel from foreign research reactors?

Response: Since the 1950's, as part of the "Atoms for Peace" program, the United States has provided peaceful nuclear technology to foreign nations in exchange for their promise not to develop nuclear weapons. A major element of this

program was the provision of research reactor technology and the highly-enriched uranium necessary to fuel the research reactors. Research reactors play a vital role in important medical, agricultural, and industrial applications. To maintain control over the highly-enriched uranium that it provided to other nations, the United States has accepted SNF from these research reactors. The United States began accepting SNF from foreign research reactors in approximately 1958, a program that continued for HEU until 1988. In 1996, DOE initiated a new acceptance policy that will be in effect for ten years, under which the United States is once again accepting SNF from eligible foreign research reactors. In addition, the United States is developing LEU fuels to replace the HEU fuels still used by some foreign research reactors, in an effort to further reduce the use of HEU in commerce worldwide. These measures, taken together, are designed to reduce the threat of nuclear proliferation.

M2-21: Does DOE emphasize long-term or short-term impacts and for how long do you evaluate long-term impacts?

Response: DOE considers both short-term and long-term impacts in making decisions regarding SNF management. As indicated in Section 2.4, impacts that would occur immediately (e.g., operation of new and existing processing facilities) receive greater weight than impacts that are not expected but could occur in the distant future.

In this EIS, DOE has analyzed the treatment and interim storage of SNF at the SRS through the year 2035. Beyond 2035, it becomes increasingly difficult to reasonably anticipate program needs. For planning purposes, DOE assumes treated SNF will be transferred to a geologic repository by 2015. The long-term impacts (i.e., thousands of years) from utilizing a geologic repository for commercial SNF and DOE SNF and high-level waste are currently being evaluated under the Yucca Mountain Repository EIS.

M2-22: Does the 50-year committed dose analysis of latent cancer fatalities consider cross-generation (genetic) effects? If they do not, why? Can these effects not be cross-generational?

Response: DOE calculates adverse health effects to workers and the general public in terms of an estimated number of total fatal cancers. Given that the calculated number of excess cancers reported in the EIS are less than one for all alternatives and that the risk of genetic effects is smaller than the latent cancer risk (on a per person-rem basis), DOE does not expect that there would be any cross-generational impacts from SNF management at SRS.

M2-23: Any decisions that DOE makes regarding the disposition of SNF should not burden residents of South Carolina with additional environmental hazards.

Response: DOE's goal is to develop and implement a safe and efficient SNF management

strategy that includes preparing aluminum-based SNF for shipment to a geologic repository.

M2-24: Certainly the use of chemical processing creates more waste than it gets rid of.

Response: Comment noted. Quantities of waste for each alternative are described in the EIS (see Table 2-10, page 2-47).

M2-25: The *Progressive Network* is not in as big of a hurry as it appears DOE is.

Response: DOE believes that this EIS covers all feasible and currently reliable methods of treating SNF and does not expect any new SNF treatment technologies to be presented in the near future. Regardless of which technology is selected, implementing the process will require several years. DOE is committed to ensuring that the health and safety vulnerabilities the fuels could pose if they remain in the wet storage basins are eliminated. See Sections 2.2 and 2.6.1.

M2-25: DOE must consider the burden of contaminated water ways that South Carolinians and Georgians have endured since the SRS began operating in the 1950s.

Response: In the EIS, DOE has analyzed the potential impact of SNF alternatives on surface water resources on and near SRS, including the Savannah River. The EIS also evaluates impacts to members of the general public who could be exposed to radioactivity in these waters from SNF management activities. Tritium was included in the inventory on which the analyses were performed. The highest estimated radiation dose from surface water releases of radioactivity is 0.057 millirem per year from the Maximum Impact Alternative, which is well below the 4 millirem per year drinking water standard.

M2-27: It is important that the results of research on the (melt-and-dilute) off gas system are made publicly available.

Response: As detailed in the response to comment L2-20, the development and review process is, in general, open to interested people from throughout the DOE system. DOE will make available, upon request, select summary reports for members of the public such as the report on off-gas system development. Contact Randy Ponik, DOE-SR (803) 952-2549 or Randall.Ponik@SRS.gov to request additional technical information on technology development activities.

M2-28: DOE should use an independent review board when evaluating the different engineering designs.

Response: As detailed in the response to comment L2-20, DOE-SR encourages full and open public review through the scoping period which preceded preparation of the Draft EIS and the public comment period on this Draft EIS. Further, DOE is currently planning for independent review of the proposed melt and dilute project.

M2-29: In the final EIS, DOE should provide more particularly quantitative, but also qualitative, details so the public can try to validate the assumptions and conclusions that are made.

Response: DOE strives to make all data and information available so that members of the public and other interested parties can independently assess the methodology, assumptions, and conclusions presented in the EIS. All reports and technical documents used to support the analysis in the EIS are available for review at the DOE public reading rooms at the Gregg-Graniteville Library of the University of South Carolina, Aiken, South Carolina, and Room 1E-190 at DOE Headquarters in the Forrestal Building in Washington, DC.

M4-01: One commenter stated that the nonproliferation report indicated that the plutonium is separated in treatment. He added that this is true for Fuel Group A, which would be processed in the canyons, but it is not true for Groups B, C, and D, which would be treated by a non-separations process.

Response: The commenter is correct.

M4-02: One commenter identified two statements in the National Academy of Sciences (NAS) report on the DOE research reactor aluminum spent fuel program. The first stated that SNF disposition is a very small nonproliferation issue compared with the nonproliferation issue in total waste management, waste disposal, and waste storage. The second stated it is not credible to say that the choice of technology the United States makes for managing this spent fuel will have any effect on nonproliferation decisions other nations will make.

Response: DOE's nonproliferation report, that is summarized and referenced in the EIS, concluded that all SNF treatment technologies, including processing, were compatible with United States nonproliferation policy.

M4-03: The commenter also stated that the NAS report also said that it is difficult to accept the nonproliferation report done by DOE as independent in such a controversial area.

Response: The nonproliferation report, *Nonproliferation Impact Assessment for the Management of The Savannah River Site Aluminum-Based Spent Nuclear Fuel*, December 1998 is referenced in this EIS and is available in the DOE public reading rooms. It was prepared by DOE's Office of Arms Control and Nonproliferation. Prior to its publication, the report was reviewed for comments by experts outside DOE. The first page of the document acknowledges the five independent reviewers. They are from Harvard, Princeton, and Stanford Universities, Bengelsdorf, McGoldrick and Associates, and the Institute for Science and International Security.

M4-04: One commenter noted that DOE's identification of decisions to be reached in the EIS included whether DOE should construct new facilities or use existing facilities to store and treat or package the spent nuclear fuel that would be managed at SRS in preparation for its ultimate disposition. The commenter asked where that decision is addressed in the EIS.

M4-05: The commenter also stated that he saw very little on whether DOE should construct a new facility or use existing facilities.

Response to comments M4-04 and -05: As discussed in the Summary, Section S.11, Preferred Alternative, to implement the preferred alternative, DOE would construct a melt and dilute facility in the existing 105-L building at SRS and build a dry-storage facility in L-Area, near the 105-L building. The section also identifies conventional processing in the existing canyons for the treatment of a relatively small volume (about 3 percent by volume and 40 percent by mass) of aluminum-based SNF. Section 2.4.3 of the EIS details components of the preferred alternative, including a new melt and dilute facility in an existing building, existing conventional processing, temporary use of existing wet-storage, and a new dry storage facility. The EIS provides an analysis of renovating an existing facility vs. constructing a new facility for SNF management (see Section 4.3.7) to allow DOE to make an informed decision regarding this issue.

M4-06: The commenter followed with three comments related to waste acceptance at the geologic repository. In comment M4-06 he stated that without a clear definition of what is road-ready (and suitable for disposal in a geologic repository) DOE should go to a proven technology (conventional processing).

M4-07: The commenter stated that the EIS does not help him understand how DOE would reach a decision on the acceptability of the waste forms that would result from application of the various technologies.

M4-08: The commenter recommended that "DOE take a strong charter" to not shut down the canyons until it has determined the waste forms that are suitable for disposal in a geologic repository

Response to Comments M4-06, -07, and -08: DOE-SR is working closely with the Nuclear Regulatory Commission (the federal agency that would license the operation of a geologic re-

pository) to ensure that the final product from the selected SNF treatment technology would be acceptable for disposition. DOE expects repository waste acceptance criteria to be established well before the melt and dilute system begins operation. The EIS has been revised to discuss in greater detail the expected repository acceptance criteria and compare the treatment technology products to those criteria. This information is discussed in Sections 2.2.1.

Recognizing that repository disposal is the ultimate endpoint for the melt and dilute waste form, DOE-SR signed in August 1997 a Memorandum of Understanding with the Nuclear Regulatory Commission (NRC) for their review and feedback on the research effort that DOE is conducting. DOE has provided the NRC several technical reports on the results obtained from the research effort. Based upon their initial review, the NRC in a June 1998 letter stated that "both the direct co-disposal and melt-dilute options would be acceptable concepts for the disposal of aluminum-based research reactor SNF in the repository." Additionally, as research efforts yield new findings, DOE is providing the information to the NRC for their feedback and review.

M4-09: The commenter also stated that DOE should "get on with it." DOE should process the SNF or have an alternative that is available and suitable for disposal in the geologic repository.

Response: DOE's proposed action is to select a new non-chemical processing technology that would put SNF into a form or container suitable for direct placement in a geologic repository. DOE's preferred alternative is to treat the majority of SNF using a non-separations-based technology, i.e., melt and dilute. DOE proposes to apply the Conventional Processing technology only to a relatively small volume of SNF at SRS (about 3 percent by volume and 40 percent by mass) before a new treatment facility is in place. The rationale for this processing is to avoid the possibility of future urgent actions, including expensive recovery actions that would entail unnecessary radiation exposure to work-

ers, and in one case, to manage a unique waste form (i.e., core filter block).

M4-10: One commenter asked if Savannah River was fully funded for the planned programs in 2000 and 2001, what was the source of money to implement new technologies, and would it take funds from existing missions.

Response: DOE has conducted laboratory testing for the melt and dilute process and plans to begin a pilot scale facility test program.

The budget rollout process that occurred on February 2, 1999, involved funding decisions based on a prioritization of DOE's goals for the upcoming years. DOE considers the development of an alternate SNF treatment technology an important part of the Site's overall mission and the program funding has been budgeted accordingly.

M4-11: One commenter noted that it did not make sense to him to develop a new technology to replace an existing one or to slow ongoing SRS operations in order to fund technology development.

Response: See response to comment M4-10.

M4-12: One commenter noted that processing costs were strongly influenced by the need to build a new facility to process the small amount of foreign fuel that would be received after 2015. He suggested an alternative that would not provide that facility so the reader could understand the cost implications of processing versus the other technologies.

Response: Based on previous decisions by the Department, DOE has assumed that there is a set amount of aluminum-based SNF inventory that will be delivered to SRS. As a result, DOE must plan to manage that inventory. DOE does not consider it a reasonable alternative to not build a new facility to deal with the fuel scheduled to be delivered to SRS after 2015.

M4-13: The commenter continued with a request to clearly identify the amount expected to

be received after 2015, its source, and its non-proliferation issues.

Response: Receipts would be about 150 Materials Test Reactor-like elements per year and 12 High Flux Isotope Reactor assemblies per year. This information has been provided in Section 2.4.5 and Section C.2.1. The nonproliferation issues for the treatment of aluminum-based SNF, including the fuel expected to be received after 2015, are discussed in Section 2.6 of the EIS.

M4-14: The commenter stated that before DOE shuts down processing it needs assurance that there is another method to render the waste suitable for disposal in the geologic repository. If there is not, plan to process it.

Response: See response to comments M4-06, -07, and -08.

M4-15: The commenter asked how foreign fuel that was originally expected but was not received affects nonproliferation.

Response: When the Foreign Research Reactor Spent Nuclear Fuel Acceptance Program ends in 2009, DOE expects to have received about 75 percent of the United States-provided highly enriched uranium that was used abroad. In addition, through the efforts of the Reduced Enrichment for Research and Test Reactors Program, many reactors worldwide will convert from HEU fuel to LEU fuel, thereby reducing the use of HEU in commerce worldwide. These accomplishments, when completed, will have reduced the threat of nuclear proliferation.

M4-16: The commenter stated that if SRS is going to conventionally process 90 percent or so of the fuel and only melt and dilute the remainder in a small facility, this hybrid of treatments should be evaluated as an alternative.

Response: The Conventional Processing alternative considered processing fuel in the manner described by the commenter. Because this method has the greatest impacts, those impacts

cover any combination of processing and another technology.

M4-17: One commenter read a statement from the NAS review of the cost study, "all of the analyses to date show that continued processing is the best, fastest, cheapest, and most certain of success of all of the alternatives considered." The commenter then stated that DOE should not allow the "somewhat tenuous" nonproliferation policy considerations to reject a technology that is clearly technically and economically the best.

Response: DOE considered a variety of factors in identifying its preferred alternative for SNF management. Among the factors considered were technology availability and technical feasibility, nonproliferation, safeguards and security, labor availability and core competency, minimum custodial care, cost, and environmental impacts. DOE's Record of Decision on SNF management will be based on a cumulative evaluation of all of these factors.

DOE proposes to apply the Conventional Processing technology to a relatively small volume of aluminum-based SNF at SRS (about 3 percent by volume and 40 percent by mass) before a new treatment facility is in place. The rationale for this processing is to avoid the possibility of future urgent actions, including expensive recovery actions that would entail unnecessary radiation exposure to workers, and in one case, to manage a unique waste form (i.e., core filter block).

M4-18: One commenter asked if the potential economic benefit of not diluting the uranium in the non-chemical separations process technologies was factored into the cost study.

Response: The cost study referenced in the EIS states that uranium credits in the range of \$110 million to \$150 million were considered in developing the cost report. However, due to the recently signed agreement between Russia and the United States that calls for the potential deferment of HEU sales, no actual value was assigned in the report for calculating life-cycle costs associated with processing activities. The

TC

EIS has been revised to clarify this point (see Section 2.6.5).

References

- Bickford, W. E., J. F. Krupa, D. L. McWhorter, M. E. Dupont, 1997, *Savannah River Site Spent Nuclear Fuel Environmental Impact Statement Engineering Data Book for Routine Releases*, WSRC-TR-97-0044, Westinghouse Savannah River Company, Aiken, South Carolina, April 8.
- DOE (U.S. Department of Energy), 1998b, *Report on the Savannah River Site Aluminum-Clad Spent Nuclear Fuel Alternative Cost Study*, Savannah River Operations Office, Aiken, South Carolina, December.
- Knapp, M. R., 1998, "Review of the Aluminum-based Research Reactor Spent Nuclear Fuel Disposition Program," letter from Nuclear Regulatory Commission Office of Nuclear Material Safety and Safeguards to J. E. Anderson, Acting Assistant Manager for Material and Facility Stabilization, U.S. Department of Energy, Savannah River Operations Office, Aiken, S.C. June 5.
- McWhorter, D. L., 1997, "SRS SNF Management EIS, SNF Processing Durations (U)," interoffice memorandum to C. B. Shedrow, Westinghouse Savannah River Company, Aiken, South Carolina.

OCI REPRESENTATION STATEMENT

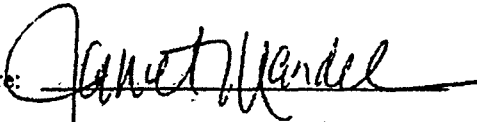
Contract No. DE-AC09-92SR18220

Task Assignment 024

Subtask 09

**National Environmental Policy Act (NEPA) Documentation, Management, and Planning Support.
Environmental Documentation/NEPA Requirements Support for Site Specific Spent Nuclear Fuel
Management Program at SRS.**

I hereby certify (or as a representative of my organization, I hereby certify) that, to the best of my knowledge and belief, no facts exist relevant to any past, present, or currently planned interest or activity (financial, contractual, personal, organizational or otherwise) which relate to the proposed work and bear on whether I have (or the organization has) a possible conflict of interest with respect to (1) being able to render impartial, technically sound, and objective assistance or advice, or (2) being given an unfair */ competitive advantage.

Signature:  Date: June 6, 1996

Name: Janet Mandel Organization: Halliburton NUS Corporation

Title: Contracting Officer

*/ An unfair competitive advantage does not include the normal flow of benefits from the performance of this contract.

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DOE is providing copies of the Final EIS to Federal, state, and local elected and appointed officials and agencies of government; Native American groups; Federal, state, and local environmental and public interest groups; and other organizations and individuals listed below. Copies will be provided to other interested parties upon request as identified in the coversheet of this Final EIS.

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GLOSSARY

°C

Degree Celsius. A temperature scale commonly used in scientific work based on the freezing point of water at 0°C and the boiling point at 100°C under normal atmospheric pressure.

$$^{\circ}\text{C} = \frac{5}{9} \times (^{\circ}\text{F} - 32).$$

°F

Degree Fahrenheit. A temperature scale based on the freezing point of water at 32°F and the boiling point at 212°F under normal atmospheric pressure. $^{\circ}\text{F} = (^{\circ}\text{C} \times \frac{9}{5}) + 32$.

absorbed dose

The energy deposited per unit mass by ionizing radiation. The unit of absorbed dose is the rad.

accident

One or more unplanned events involving materials that have the potential to endanger the health and safety of workers and the public. An accident can involve a combined release of energy and hazardous materials (radiological or chemical) that might cause prompt or latent adverse health effects.

actinide

Any of a series of chemically similar, mostly synthetic, radioactive elements with atomic numbers ranging from actinium at 89 through lawrencium at 103.

air quality standards

The prescribed level of constituents in the outside air (ambient air) that legally should not be exceeded during a specified unit of time in a specified area.

alpha (α) particle

A positively charged particle consisting of two protons and two neutrons that is emitted from the nucleus of certain nuclides during radioactive decay. It is the least penetrating of the three common types of radiation (alpha, beta, and gamma).

aqueous

Relating to or made with water.

aquifer

A geologic formation that contains enough saturated porous material to permit groundwater to move through it and to yield worthwhile quantities of groundwater to wells and springs.

As Low As Reasonably Achievable (ALARA)

An approach to radiation protection that controls or manages exposures (both individual and collective to workers and general public) as low as social, technical, economic, practical and public policy considerations permit. ALARA is not a dose limit, but a process which has the objective of dose levels as far below applicable limits of 10 CFR 835 as is reasonably achievable. Particular attention is to be paid to this definition in design of facilities.

atomic weight

The relative weight of an atom of a chemical element based on the weight of the most abundant isotope of carbon, which is taken to be 12.

AXAIRQ

A computer model that analyzes doses from airborne radionuclide releases.

background exposure

See exposure to radiation.

background radiation

Normal radiation present in the lower atmosphere from cosmic rays and earth sources. Background radiation varies with location, depending on altitude and natural radioactivity present in the surrounding geology.

Best Management Practices (BMP)

A practice or combination of practices that is determined by a state (or other planning agency) to be the most effective, practicable means of preventing pollution generated by nonpoint sources or of reducing it to a level compatible with air or water quality goals.

beta (β) particle

An elementary particle emitted from a nucleus during radioactive decay. It is negatively charged, identical to an electron, and easily stopped by a thin sheet of metal.

biota

Living organisms.

blackwater

Water in Coastal Plain creeks, swamps, and rivers that has a dark or black coloration due to dissolution of naturally-occurring organic matter from soils and decaying vegetation.

bounding accident

An accident whose calculated consequences encompass all other possible accident consequences for that facility. For example, a bounding accident for the release of hazardous material from a storage tank would postulate the release of the entire tank contents. The consequences from this accident would be greater than the consequences of all other tank release accidents.

brownfield

An area that has been previously disturbed by industrial activities.

burn

Irradiation of fuel in a nuclear reactor with the resultant production of energy, neutrons, and fission products.

burnup

The total energy released through fission by a given amount of nuclear fuel; generally measured in megawatt-days.

cancer

A malignant tumor of potentially unlimited growth, capable of invading surrounding tissue or spreading to other parts of the body by metastasis.

canister

A stainless-steel container in which nuclear material is sealed.

canyon

A heavily shielded building where radioactive materials are chemically processed to recover special isotopes for national defense or other programmatic purposes. In the canyon, operation and maintenance are remotely-controlled.

capable (geology)

Describes a geological fault that has moved at or near the ground surface within the past 35,000 years.

carcinogen

An agent capable of inducing cancer.

carcinogenic

Capable of inducing cancer.

case

The application of a given technology to a single fuel group.

cask

A massive, heavily-shielded container for holding nuclear materials during shipment.

cesium

Naturally occurring element with 55 protons in its nucleus. Some manmade isotopes of cesium are radioactive (e.g., cesium-134, cesium-137).

cladding

The outer jacket of fuel elements and targets, usually made of aluminum, stainless steel, or zirconium-aluminum alloy; used to prevent fuel corrosion and retain fission products during reactor operation, or to prevent radioactive releases into the environment during storage.

co-disposal

A disposal approach for spent nuclear fuel in a geologic repository. Five canisters of high-level waste would fit in a repository waste package, with room for one 17-inch (43-centimeter) diameter canister of spent nuclear fuel.

collective dose

The sum of the individual doses to all members of a specific population.

committed dose equivalent

The calculated dose equivalent received by a tissue or an organ during the 50-year period after a radionuclide is introduced into the body.

committed effective dose equivalent

The sum of the committed dose equivalents to various tissues/organs in the body multiplied by their appropriate tissue weighting factor. Equivalent in effect to a uniform external dose of the same value.

community (environmental justice)

A group of people or a specific location exposed to risks that potentially threaten health, ecology, or land values, or exposed to industry that stimulates unwanted noise, smell, industrial traffic, particulate matter, or other nonaesthetic impacts.

concentration

The amount of a substance contained in a unit quantity of material.

confining unit

A body of impermeable or distinctly less-permeable material stratigraphically adjacent to one or more aquifers.

consequence

The result or effect (especially projected exposure to radiological or chemical hazards) of a release of hazardous materials to the environment.

constituents

Parts or components of a whole.

critical

Describing the condition when fissile materials exposed to neutron bombardment produce enough neutrons to support a chain reaction.

criticality

A state in which a self-sustaining nuclear chain reaction is achieved.

crop

A process that cuts off or otherwise removes the hardware on the fuel assemblies, leaving primarily the active fuel for subsequent processes.

cumulative impacts

Additive impacts on the environment including ecological, human health, or socioeconomic effects that result from the addition of the impact of the proposed action to impacts from other past, present, and reasonably foreseeable future actions regardless of what agency (Federal or non-Federal) or person undertakes the other actions (40 CFR 1508.7).

curie (Ci)

A unit of radioactivity equal to 37,000,000,000 disintegrations per second (or becquerels).

daughter

A nuclide formed by the radioactive decay of another nuclide, which is the "parent."

decay heat

The radioactive decay of fission products can produce very high temperatures (decay heat), which is why fuel recently removed from a reactor is placed in underwater storage for cooling. Without active cooling, the fuel could overheat and melt or damage the cladding. After a sufficient cooling time that depends on the burnup of the fuel and its composition, fuel assemblies can be stored dry. Dry fuel storage technologies must consider decay heat.

decay, radioactive

The spontaneous transformation of one nuclide into a different nuclide or into a different energy state of the same nuclide. The process results in the emission of nuclear radiation (usually alpha, beta, or gamma radiation).

decibel

A unit for measuring the relative loudness of sounds. In general, a sound doubles in loudness for every increase of 10 decibels.

decision maker

Group or individual responsible for making a decision on a particular proposed action. Decision makers include DOE officials as specified in DOE Order 451.1A; elected officials; Federal, state, and local agency representatives; and the public.

decommissioning

The removal from service of facilities such as processing plants, waste tanks, and burial grounds, and the reduction or stabilization of radioactive contamination. Decommissioning includes decontamination, dismantling, and return of the area to original condition without restrictions or partial decontamination, isolation of remaining residues, and continuation of surveillance and restrictions.

Defense Waste Processing Facility

Savannah River Site facility that processes high-level radioactive waste into a glass form for transport to a permanent disposal site.

deflagration

Rapid burning with great heat and intense light.

demographic

Related to the statistical study of human populations, including size, density, distribution, and vital statistics such as age, gender, and ethnicity.

depleted uranium

A mixture of uranium isotopes where uranium-235 represents less than 0.7 percent of the uranium by mass.

derived concentration guide (DCG)

The concentration of a radionuclide in air or water that, under conditions of continuous exposure for one year by one exposure mode (i.e., ingestion of water, submersion in air, or inhalation), would result in an effective dose equivalent of 100 mrem (0.1 rem = 1 mSv [milliSievert]).

disassociate

Separate chemicals into their elemental or ionic state.

dose

The energy imparted to matter by ionizing radiation. The unit of absorbed dose is the rad, equal to 0.01 joule per kilogram of irradiated material in any medium.

dose conversion factor

Factor used to calculate the dose received from exposure to radiation.

dose equivalent

A term used to express the amount of effective radiation when modifying factors have been considered. It is the product of absorbed dose (rads) multiplied by a quality factor and other modifying factors. It is measured in rem (*Roentgen equivalent man*).

dose rate

The radiation dose delivered per unit time (e.g., rem per year).

ecology

The science that deals with the relationship of living things with each other and with their environment.

ecosystem

A complex of the communities of living things and their environment which forms a functioning whole in nature.

effective dose equivalent

A quantity used to estimate the biological effect of ionizing radiation. It is the sum over all body tissues of the product of absorbed dose, the quality factor (to account for the different penetrating abilities of the various radiations), and the tissue weighting factor (to account for the different radiosensitivity of the various tissues of the body).

effective porosity

A property of earth containing interconnecting interstices, expressed as a percent of bulk volume occupied by the interstices.

effluent

Liquid or airborne material released to the environment. In general usage, however, effluent implies liquid releases.

electron

An elementary particle with a mass of 9.107×10^{-28} gram (or 1/1837 of a proton) and a negative charge. Electrons surround the positively charged nucleus and determine the chemical properties of the atom.

element

One of the 109 known chemical substances that cannot be divided into simpler substances by chemical means. All isotopes of an element have the same atomic number (number of protons) but have different numbers of neutrons.

Emergency Response Planning Guideline (ERPG) values

These values, which are specific for each chemical, are established for three general severity levels: exposure to concentrations greater than ERPG-1 values for a period of time greater than

1 hour results in an unacceptable likelihood that a person would experience mild transient adverse health effects, or perception of a clearly defined objectionable odor; exposure to concentrations greater than ERPG-2 values for a period of time greater than 1 hour results in an unacceptable likelihood that a person would experience or develop irreversible or other serious health effects, or symptoms that could impair one's ability to take protective action; exposure to concentrations greater than ERPG-3 values for a period of time greater than 1 hour results in an unacceptable likelihood that a person would experience or develop life-threatening health effects.

emission standards

Legally enforceable limits on the quantities and kinds of air contaminants that may be emitted to the atmosphere.

endangered species

Plants or animals that are in danger of extinction through all or a significant portion of their ranges and that have been listed as endangered by the U.S. Fish and Wildlife Service or the National Marine Fisheries Service.

energy

The capacity to produce heat or do work.

enrichment

A process in which the fraction of the uranium-235 isotope has been artificially increased above the natural abundance level of 0.72 percent.

environment

The sum of all external conditions and influences affecting the life, development, and ultimately the survival of an organism.

environmental impact statement (EIS)

A detailed written statement as required by Section 102(2)(c) of the National Environmental Policy Act (NEPA) of 1969, as amended, to assess the environmental impacts of major Federal actions.

environmental justice

The fair treatment of people of all races, cultures, incomes, and educational levels with respect to the development, implementation, and enforcement of environmental laws, regulations, and policies. Fair treatment implies that no population of people should be forced to shoulder a disproportionate share of the negative environmental impacts of pollution or environmental hazards due to a lack of political or economic strength.

exposure to radiation

The incidence of radiation on living or inanimate material by accident or intent. Background exposure is the exposure to natural background ionizing radiation. Occupational exposure is the exposure to ionizing radiation that occurs at a person's workplace. Population exposure is the exposure to a number of persons who inhabit an area.

external initiators

Accidental occurrences that are independent of facility operations and normally originate outside the facility (aircraft crashes, nearby explosions, and toxic chemical releases at nearby facilities).

that affect worker performance); some can affect the ability of the facility to maintain confinement of hazardous materials because of structural damage.

fault

A fracture or a zone of fractures within a rock formation along which vertical, horizontal, or transverse slippage of the earth's crust has occurred in the past.

fertile

Describing radionuclides that can be converted into fissile material (e.g., thorium-232 and uranium-238 can be converted through neutron capture to uranium-233 and plutonium-239, respectively).

fissile

Capable of being split or divided (fissioned) by the absorption of thermal neutrons. The most common fissile materials are uranium-233, uranium-235, and plutonium-239.

fission

The splitting of a heavy nucleus into two approximately equal parts, which are nuclei of lighter elements, accompanied by the release of energy and generally two or more neutrons. Fission can occur spontaneously or can be induced by nuclear bombardment.

fission chain reaction

Nuclear reaction in which atomic nuclei in reactor fuel respond to collisions with neutrons by splitting into two or three major fragments and additional neutrons accompanied by the emission of gamma radiation.

fission fragments

The parts into which atomic nuclei in reactor fuel split during a fission chain reaction.

fission products

Nuclei from the fission of heavy elements (primary fission products); also, the nuclei formed by the decay of the primary fission products, many of which are radioactive.

gamma (γ) rays

High-energy, short-wavelength electromagnetic radiation accompanying fission, radioactive decay, or nuclear reactions. Gamma rays are very penetrating and require relatively thick shields to absorb the rays effectively.

geology

The science that deals with the earth: the materials, processes, environments, and history of the planet, especially the lithosphere, including the rocks, their formation and structure.

groundwater

The supply of fresh water below the earth's surface in an aquifer.

habitat

The place or type of site where a plant or animal normally grows or lives.

half-life (radiological)

The time in which half the atoms of a radioactive substance disintegrate to another nuclear form. Half-lives vary from millionths of a second to billions of years.

hazardous material

A substance or a material including a hazardous substance that has been determined by the U.S. Secretary of Transportation to be capable of posing an unreasonable risk to health, safety, and property when transported in commerce.

hazardous substance

Any substance that when released to the environment in an uncontrolled fashion could be harmful to the biota or human health and when released in an unpermitted fashion becomes subject to the reporting and possible response provisions of the Clean Water Act and the Comprehensive Environmental Response, Compensation, and Liability Act.

hazardous waste

Waste that is regulated under the Resource Conservation and Recovery Act and corresponding state regulations. Waste is hazardous if the EPA lists it as such or if it exhibits the characteristic(s) of ignitability, corrosivity, reactivity, or toxicity. SRS hazardous waste streams consist of a variety of materials, including mercury, chromates, lead, paint solvents, and various laboratory chemicals.

heavy metal

In this document, heavy metal refers to materials of high atomic number that were placed in nuclear reactors. This includes thorium, uranium, and plutonium.

high efficiency particulate air (HEPA) filter

A type of filter designed to remove 99.95 percent of the particles down to 0.3 micrometer in diameter from a flowing air stream.

high(ly) enriched uranium

Uranium that is equal to or greater than 20 percent uranium-235 by weight. Many of the fuels discussed in this EIS are based primarily on highly enriched uranium.

high-level radioactive waste

Highly radioactive material from the processing of spent nuclear fuel that contains a combination of transuranic waste and fission products in concentrations that require permanent isolation. It includes both liquid waste produced by processing and solid waste derived from that liquid.

hydraulic conductivity

The rate of water flow in gallons per day through a cross-section of 1 square foot under a unit hydraulic gradient, also known as permeability coefficient.

hydraulic gradient

With regard to an aquifer, the rate of change of pressure head per unit distance of flow at a given point and in a given direction.

infrastructure

The system of public works of a county, state, or region; also, the resources (buildings or equipment) required for an activity.

interim storage

Safe and secure storage for spent nuclear fuel and radioactive wastes until the materials are positioned (treatment and/or disposal).

internal initiators

Events that normally originate in and around the facility but are always a result of facility operations (equipment or structural failures, human errors, internal flooding). In accident scenarios, initiators start the events that culminate in a release of hazardous or radioactive materials.

involved worker

An individual located in the facility under discussion.

ion

An atom or molecule that has gained or lost one or more electrons to become electrically charged.

ionizing radiation

Radiation capable of ejecting electrons from atoms or molecules to produce ions.

irradiation

Exposure to radiation.

isotope

An atom of a chemical element with a specific atomic number and atomic weight. Isotopes of the same element have the same number of protons but different numbers of neutrons (i.e., the same atomic number, but different mass numbers). Isotopes are identified by the name of the element and the total number of protons and neutrons in the nucleus. For example, plutonium-239 is a plutonium atom with 239 protons and neutrons.

isotopic dilution

Mixing a less-enriched radioisotope with a highly enriched radioisotope to yield an isotope with lower nuclear enrichment.

latent cancer fatalities

Deaths resulting from cancers that became active sometime after the exposure presumed to have induced them.

long-lived radionuclides

Radioactive isotopes with half-lives greater than about 30 years.

low-enriched uranium (LEU)

Uranium with uranium-235 enriched above the natural concentration (0.72 percent) but below 20 percent; highly enriched uranium (HEU) is enriched 20 percent or higher.

low-income community

A community where 25 percent or more of the population is identified as living in poverty.

low-level mixed waste

Radioactive waste that contains material listed as hazardous under the Resource Conservation and Recovery Act or that exhibits one or more of the following hazardous waste characteristics: ignitability, corrosivity, reactivity, or toxicity. It includes such materials as tritiated mercury, tritiated oil contaminated with mercury, other mercury-contaminated compounds, or radioactively-contaminated lead shielding.

low-level radioactive waste

Radioactive waste that cannot be classified as high-level waste, spent nuclear fuel, transuranic waste, or byproduct material, and that does not have any constituents that are regulated under the Resource Conservation and Recovery Act.

materials test reactor equivalent (MTRE)

A quantity of spent nuclear fuel related to its volume that provides information on the amount of storage space provided.

MAXIGASP

A computer program used to calculate doses of airborne releases of radioactivity to the maximally exposed member of the public.

maximally exposed individual

A hypothetical person located to receive the maximum possible dose by a given exposure scenario.

maximum contaminant levels (MCLs)

The maximum permissible level of a contaminant in water that is delivered to a user of a public water system.

metric tons of heavy metal (MTHM)

Quantities of unirradiated and spent nuclear fuel and targets are traditionally expressed in terms of metric tons of heavy metal (typically uranium) without the inclusion of other materials such as cladding, alloy materials, and structural materials. A metric ton is 1,000 kilograms, which is equal to about 2,200 pounds.

migration

The natural travel of a material through the air, soil, or water.

millirem

One thousandth of a rem. (See rem.)

minority community

A person classified by the U.S. Bureau of the Census as Black, Hispanic, Asian and Pacific Islander, American Indian, Eskimo, Aleut, or other nonwhite persons is considered a minority. A community with the number of minority persons equal to or greater than the minority average of a defined area or jurisdiction (usually a state) is a minority community.

moderation

Process for slowing down neutrons resulting from fission or other nuclear reactions; slow or "thermal" neutrons are necessary for sustaining a fission chain reaction in fissile materials; water and heavy water are common moderators.

monitoring

Continuing control and accountability, particularly of special nuclear materials such as plutonium-239 and highly enriched uranium, but also including oversight of hazardous or reactive compounds before they are disposed of or converted to a stable long-term storage form.

National Ambient Air Quality Standards

Air quality standards established by the Clean Air Act, as amended in 1990. The primary National Ambient Air Quality Standards are intended to provide the public with an adequate margin of safety, and the secondary National Ambient Air Quality Standards are intended to protect the public from known or anticipated adverse impacts of a pollutant.

National Pollutant Discharge Elimination System

Federal system that permits liquid effluents regulated through the Clean Water Act, as amended.

natural phenomena initiators

Natural occurrences that are independent of facility operations and events at nearby facilities or operations (earthquakes, high winds, floods, lightning, snow). Although these initiators are independent of external facilities, they can affect such facilities and compound the progression of the accident.

natural radiation or natural radioactivity

Background radiation. Radiation arising from cosmic and terrestrial naturally-occurring radionuclide sources.

National Environmental Policy Act (NEPA)

The National Environmental Policy Act of 1969 (42 USC 4321) requires the preparation of an EIS for Federal projects that could incur significant impacts to the environment.

neutron

An elementary nuclear particle capable of inducing a fission chain reaction in certain atomic nuclei, including uranium-235.

neutron poison

A substance that absorbs neutrons without causing a fission, thereby preventing nuclear criticalities.

noninvolved worker

For this EIS, an SRS worker who is not involved in a given operation or activity.

nonproliferation

The restriction of access to fissile materials in concentrations sufficient to assemble a nuclear weapon.

Nuclear Regulatory Commission

The independent Federal commission that licenses and regulates commercial nuclear facilities.

nuclear radiation

Radiation, usually alpha, beta, or gamma, that emanates from an unstable atomic nucleus.

nuclear reaction

An interaction between a photon, particle, or nucleus and a target nucleus, leading to transmutation.

nuclear reactor

A device in which a fission chain reaction is maintained, used for the irradiation of materials or the generation of electricity.

nuclide

A species of atom characterized by the number of protons, number of neutrons and by energy content in the nucleus; a radionuclide is a radioactive nuclide.

offsite population

Defined as all individuals located within an 80-kilometer (50-mile) radius of a facility with potential to emit radioactive material.

organic compounds

Chemical compounds containing hydrocarbons.

ozone

A compound of oxygen in which three oxygen atoms are chemically attached to each other.

oxides of nitrogen (NO_x)

Primarily nitrogen oxide (NO) and nitrogen dioxide (NO₂), these compounds are produced in the combustion of fossil fuels, and contribute to air pollution.

particulates

Solid particles and liquid droplets small enough to become airborne.

passivation

To reduce the reactivity of a chemically-active metal.

pellets

One configuration of the reactive material in a target rod.

permeability

A measure of a material's ability to have liquids or gases pass through it via pores or openings.

person-rem

The radiation dose to a given population; the sum of the individual doses received by a population.

plutonium (Pu)

A transuranic, heavy (average atomic mass about 244 atomic mass units), silvery metal with 15 known isotopes that is produced by the neutron irradiation of natural uranium. Plutonium-239 is used both in nuclear weapons and commercial nuclear power applications. Plutonium-238 is used to power onboard electric generators during manned and unmanned space flights.

poison

A material that has an affinity for absorbing neutrons. Poisons are added to nuclear materials with a potential criticality concern to lessen the likelihood of an uncontrolled nuclear reaction.

pollution

The addition of an undesirable agent to an ecosystem in excess of the rate at which natural processes can degrade, assimilate, or disperse it.

POPGASP

A computer model used to calculate doses of airborne releases of radioactivity to the population within 80 kilometers (50 miles) of the SRS.

population

In this EIS, a collection of members of the public that is located outside the boundaries of the SRS. Impacts in this EIS are estimated for the population within a given area, depending on the appropriate environmental pathways. For example, the affected population for liquid releases to the Savannah River includes downstream residents.

Prevention of Significant Deterioration (PSD)

A standard that establishes the acceptable amount of deterioration in air quality. When the air quality of an area meets the standards for a specific pollutant, the area is declared to be in attainment for that pollutant. When the air quality of an area does not meet the standard for a specific pollutant, the area is said to be in nonattainment for that pollutant. PSD requirements define maximum allowable increases (increments) in ambient air pollutant concentrations (sulfur dioxide, particulate, nitrogen oxide) for construction or modification of facilities which by definition do not "significantly deteriorate" the existing baseline air quality.

processing (of spent nuclear fuel)

Applying a chemical or physical process designed to alter the characteristics of the spent fuel matrix.

production well/water

At the SRS, water treated and used as potable water.

programmatic materials

Stable nuclear materials with value for supporting national programs (e.g., plutonium-238 production for the National Aeronautics and Space Administration).

proton

A nuclear particle with a positive charge equal in magnitude to the negative charge of the electron; it is a constituent of all atomic nuclei, and the atomic number of an element indicates the number of protons in the nucleus of each atom of that element.

pyrophoric

The tendency to spontaneously ignite in air. Some uranium and thorium metal fuels may be pyrophoric.

radiation

The emitted particles and photons from the nuclei of radioactive atoms; a shortened term for ionizing radiation or nuclear radiation as distinguished from nonionizing radiation (microwaves, ultraviolet rays, etc.).

radiation shielding

Radiation-absorbing material that is interposed between a source of radiation and organisms that would be harmed by the radiation (e.g., people).

radioactive waste

Waste that has radioactive material and must be handled as such.

radioactivity

The spontaneous decay of unstable atomic nuclei, accompanied by the emission of radiation.

radioisotope

Radioactive isotopes. The isotopes of an element that are radioactive. Not all isotopes of a single element are radioactive. Some radioisotopes are naturally occurring (e.g., potassium-40) while others are produced by nuclear reactions.

radiolysis

Decomposition of a material by ionizing radiation.

radionuclide

A nuclide that exhibits radioactivity.

reactor

A device in which a chain reaction of fissionable material is initiated and controlled; a nuclear reactor.

Record of Decision (ROD)

A document that provides a concise public record of an agency decision on a proposed action described in an EIS. An ROD identifies the alternatives, the environmentally preferable alternative(s), factors the agency considered in making the decision, and whether the agency has adopted all practicable means to avoid or minimize environmental harm and if not, why not.

recycling

Return of a waste material either to the process that generated the waste or to another process to use or reuse the waste material beneficially; recovery of a useful or valuable material from waste.

rem

The unit of radiation dose for biological absorption. It is equal to the product of the absorbed dose in rads, a quality factor and a distribution factor.

remote handling cell

A room designed so that the process carried out in the room is done remotely by operators manipulating robotic equipment.

repository

A place for the disposal of immobilized high-level waste and spent nuclear fuel in isolation from the environment.

processing (of spent nuclear fuel)

Processing of reactor-irradiated nuclear material (primarily spent nuclear fuel) to recover fissile and fertile material, in order to recycle such materials primarily for defense programs or generation of electricity. Historically, processing has involved aqueous chemical separations of elements (typically uranium or plutonium) from undesired elements in the fuel.

resin

An ion-exchange medium; organic polymer used for the preferential removal of certain ions from a solution.

Richter Scale

A scale to quantify earthquake intensity.

risk

In accident analysis, the probability-weighted consequence of an accident, defined as the accident frequency per year multiplied by the dose. The term "risk" also is used commonly in other applications to describe the probability of an event occurring.

road-ready

Describing spent nuclear fuel that has been conditioned or treated and placed in a canister in a form such that it can be shipped to a repository.

runoff

The portion of rainfall, melted snow, or irrigation water that flows across ground surface and eventually returns to streams. Runoff can carry pollutants into receiving waters.

saltstone

Low-radioactivity fraction of high-level waste mixed with cement, flyash, and slag to form a concrete matrix.

sanitary waste

Solid waste that is neither hazardous, as defined by the Resource Conservation and Recovery Act, nor radioactive. It consists of salvageable material and material that is suitable for disposition in a municipal sanitary landfill. Sanitary waste streams include such items as paper, glass, discarded office material, and construction debris.

seismicity

The tendency for earthquakes to occur.

shielded transport casks

A heavily shielded container designed to hold one or more fuel elements during transport.

short-lived

A designation for radionuclides with relatively short half-lives (i.e., they decay to other atoms relatively quickly). Radionuclides with half-lives less than approximately 30 years are short-lived.

spent nuclear fuel

Fuel and targets that have been irradiated in a nuclear reactor. Spent nuclear fuel is highly radioactive.

stabilization

The action of making a nuclear material more chemically or physically stable by converting its physical or chemical form or placing it in a more stable environment.

strontium

Naturally occurring element with 38 protons in its nucleus. Some manmade isotopes of strontium are radioactive (e.g., strontium-89, strontium-90).

sulfur dioxide

A heavy, pungent, toxic gas, used as a preservative or refrigerant, that is an air pollutant.

surface water

Water on the surface (streams, ponds, etc.), as distinguished from underground water.

tank farm

An installation of interconnected underground tanks for the storage of high-level radioactive liquid wastes.

target

In this EIS, a tube of material placed in a reactor for bombardment by neutrons to produce desired radioactive byproducts.

thermal neutrons

Neutrons that have had excess energy removed by scattering collisions with other atoms and will not slow down any further. Thermal neutrons have an energy of 0.025 electron-volts and are readily absorbed by fissile atoms.

threatened species

Any species which is likely to become endangered within the foreseen future throughout all or a significant portion of its range, and that has been listed as threatened by the U.S. Fish and Wildlife Service or the National Marine Fisheries Service.

transmutation

The conversion of one element to another by means of a nuclear reaction.

transuranic waste

Waste material containing more than a specified concentration of transuranic elements (elements with higher atomic numbers than uranium) (presently, more than 10 nanocuries per gram of waste).

tritium

A radioactive isotope of hydrogen; its nucleus contains one proton and two neutrons.

uranium (U)

A heavy (average atomic mass of about 238 atomic mass units), silvery-white metal with 14 radioactive isotopes. One of the isotopes, uranium-235, is most commonly used as fuel for nuclear fission and another, uranium-238, is transformed into fissionable plutonium-239 following its capture of a neutron in a nuclear reactor.

vault

A reinforced concrete structure for storing strategic nuclear materials used in national defense or other programmatic purposes or for disposing of radioactive or hazardous waste.

vitrification

Immobilization by incorporating into glass.

vulnerability

Condition or weakness that could lead to exposure to radioactive elements by the public, unnecessary or increased exposure to workers, or release of radioactive materials to the environment.

Waste Isolation Pilot Plant (WIPP)

DOE facility located near Carlsbad, New Mexico, for the safe underground disposal of transuranic waste from numerous facilities owned by DOE.

waste minimization

Reduction of waste before treatment, storage, or disposal by source reduction or recycling activities.

waste, radioactive

See "radioactive waste."

water quality standard

Provisions of state or Federal law that consist of a designated use or uses for the waters of the United States and water quality criteria for such waters based upon those uses. Water quality standards are used to protect the public health or welfare, enhance the quality of water, and serve the purposes of the Act.

Staff Attorney, Entergy Power Marketing Corp., 10055 Grogan's Mill Road, Suite 500, The Woodlands, TX 77380.

Comments on Solutions' request to export should be clearly marked with Docket EA-155-A. Additional copies are to be filed directly with:

Richard Staines, Consolidated Edison Solutions, Inc., 701 Westchester Avenue, Suite 320E, White Plains, NY 10604; and

Steven J. Ross, Steptoe & Johnson, LLP, 1330 Connecticut Avenue, NW, Washington, DC 20036.

Comments on DETM's request to export should be clearly marked with Docket EA-163-A. Additional copies are to be filed directly with:

Kris Errickson, Legal/Regulatory Coordinator, Duke Energy Trading and Marketing, One Westchase Center, 10777 Westheimer Street, Suite 650, Houston, TX 77042;

Christine M. Pallenik, Managing Counsel, Duke Energy Trading and Marketing, 4 Triad Center, Suite 1000, Salt Lake City, UT 84180; and

Gordon J. Smith, Esq., John & Hengerer, 1200 17th Street, NW, Suite 600, Washington, DC 20036.

Comments on ComEd's request to export should be clearly marked with Docket EA-169-A. Additional copies are to be filed directly with:

Peter Thornton, Esq., Senior Counsel, Commonwealth Edison Company, 125 South Clark Street, Room 1535, Chicago, IL 60603; and

James H. McGrew, Esq., Bruder, Gentile & Marcoux, 1100 New York Avenue, NW, Suite 510 East, Washington, DC 20005-3934.

A final decision will be made on these applications after the environmental impacts have been evaluated pursuant to the National Environmental Policy Act of 1969 and determinations are made by the DOE that the proposed actions will not adversely impact on the reliability of the U.S. electric power supply system.

Copies of these applications will be made available, upon request, for public inspection and copying at the address provided above or by accessing the Fossil Energy Home Page at <http://www.fe.doe.gov>. Upon reaching the Fossil Energy Home page, select "Electricity" from the "Regulatory Info" menu, and then "Pending Proceedings" from the options menus.

Issued in Washington, DC, on January 5, 2000.

Anthony J. Como,

Deputy Director, Electric Power Regulation, Office of Coal & Power Im/Ex, Office of Coal & Power Systems, Office of Fossil Energy.

[FR Doc. 00-592 Filed 1-10-00; 8:45 am]

BILLING CODE 6450-01-P

DEPARTMENT OF ENERGY

Record of Decision for the Surplus Plutonium Disposition Final Environmental Impact Statement

AGENCY: Department of Energy.

ACTION: Record of decision.

SUMMARY: In November 1999, the Department of Energy (DOE or the Department), in accordance with the National Environmental Policy Act (NEPA), issued the Surplus Plutonium Disposition Final Environmental Impact Statement (SPD EIS)(DOE/EIS-0283). The SPD EIS was the culmination of a process started on May 22, 1997, when DOE published a Notice of Intent (NOI) in the Federal Register (62 FR 28009) announcing its decision to prepare an EIS that would tier from the analysis and decisions reached in connection with the Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic EIS (Storage and Disposition PEIS)(DOE/EIS-0229). Accordingly, the Surplus Plutonium Disposition Draft Environmental Impact Statement (SPD Draft EIS) (DOE/EIS-0283-D) was prepared and issued in July 1998. It identified the potential environmental impacts of reasonable alternatives for the proposed siting, construction, and operation of three facilities for the disposition of up to 50 metric tons of surplus plutonium, as well as a No Action Alternative. These three facilities would accomplish pit¹ disassembly and conversion, plutonium conversion and immobilization, and mixed oxide (MOX)² fuel fabrication. The SPD Draft EIS also analyzed the potential impacts of fabricating a limited number of MOX fuel assemblies, referred to as lead assemblies, for testing in a reactor before starting full production of MOX fuel, and the potential impacts of examining the lead assemblies after irradiation.

For the alternatives that included MOX fuel fabrication, the SPD Draft EIS described the potential environmental impacts of using from three to eight commercial nuclear reactors to irradiate MOX fuel. The potential impacts were

¹ A nuclear weapon component.

² A physical blend of uranium oxide and plutonium oxide.

based on a generic reactor analysis included in the Storage and Disposition PEIS that used actual reactor data and a range of potential site conditions. In May 1998, DOE initiated a procurement process to obtain MOX fuel fabrication and reactor irradiation services. In March 1999, DOE awarded a contract to Duke Engineering & Services, COGEMA Inc., and Stone & Webster (known as DCS) to provide the requested services. Full implementation of the base contract was contingent upon the successful completion of the NEPA process. A Supplement to the SPD Draft EIS (DOE/EIS-0283-S) was issued in April 1999, which analyzed the potential environmental impacts of using MOX fuel in six specific reactors named in the DCS proposal. Those reactors are: Catawba Nuclear Station Units 1 and 2 in South Carolina, McGuire Nuclear Station Units 1 and 2 in North Carolina, and North Anna Power Station Units 1 and 2 in Virginia. The SPD Final EIS addresses the comments received during the public review process for the SPD Draft EIS and the Supplement to the draft.

The Department has decided to implement a program to provide for the safe and secure disposition of up to 50 metric tons of surplus plutonium as specified in the Preferred Alternative in the Surplus Plutonium Disposition Final Environmental Impact Statement. The fundamental purpose of the program is to ensure that plutonium produced for nuclear weapons and declared excess to national security needs (now and in the future) is never again used for nuclear weapons. Specifically, the Department has decided to use a hybrid approach for the disposition of surplus plutonium. This approach allows for the immobilization of approximately 17 metric tons of surplus plutonium and the use of up to 33 metric tons of surplus plutonium as MOX fuel. The Department has selected the Savannah River Site in South Carolina as the location for all three disposition facilities. Based upon this selection, the Department will authorize DCS to fully implement the base contract. In addition, the Department has selected the Los Alamos National Laboratory in New Mexico as the location for lead assembly fabrication and Oak Ridge National Laboratory in Tennessee as the site for post-irradiation examination of lead assemblies.

As previously stated in the Storage and Disposition PEIS Record of Decision (62 FR 3014, January 21, 1997), the use of MOX fuel in existing reactors will be undertaken in a manner that is consistent with the United States' policy objective on the irreversibility of the

nuclear disarmament process and the United States' policy discouraging the civilian use of plutonium. To this end, implementing the MOX alternative will include government ownership and control of the MOX fuel fabrication facility at a DOE site, and use of the facility only for the surplus plutonium disposition program. There will be no reprocessing or subsequent reuse of spent MOX fuel. The MOX fuel will be used in a once-through fuel cycle in existing reactors, with appropriate arrangements, including contractual or licensing provisions, limiting use of MOX fuel to surplus plutonium disposition.

EFFECTIVE DATE: The decisions set forth in this Record of Decision are effective upon publication of this document, in accordance with DOE's National Environmental Policy Act Implementing Procedures and Guidelines (10 CFR Part 1021) and the Council on Environmental Quality regulations implementing NEPA (40 CFR Parts 1500-1508).

ADDRESSES: Copies of the SPD EIS and this Record of Decision may be obtained by placing a call to an answering machine or facsimile machine at a toll free number (1-800-820-5156), or by mailing a request to: Bert Stevenson, NEPA Compliance Officer, Office of Fissile Materials Disposition, U.S. Department of Energy, Post Office Box 23786, Washington, DC 20026-3786.

The full SPD EIS, including the 54-page Summary, and this Record of Decision are available on the Office of Fissile Materials Disposition's web site. The address is <http://www.doe-md.com>. The full SPD EIS is also available on DOE's NEPA web site at <http://tis.ch.doe.gov/nepa>.

FOR FURTHER INFORMATION CONTACT: Questions concerning the plutonium disposition program can be submitted by calling or faxing them to the same toll free number (1-800-820-5156), or by mailing them to Mr. Bert Stevenson at the above address. Comments may also be submitted electronically by using the Office of Fissile Materials Disposition's web site. The address is <http://www.doe-md.com>.

For general information on the DOE NEPA process, please contact: Carol Borgstrom, Director, Office of NEPA Policy and Assistance, U.S. Department of Energy, 1000 Independence Avenue, S.W., Washington, DC 20585, 202-586-4600 or 1-800-472-2756.

SUPPLEMENTARY INFORMATION:

Background

The United States and Russia are working together to reduce the threat of

nuclear weapons proliferation worldwide by disposing of surplus plutonium in a safe, secure, environmentally acceptable and timely manner. Comprehensive disposition actions are needed to ensure that surplus plutonium is converted to proliferation-resistant forms. In September 1993, President Clinton issued the Non-proliferation and Export Control Policy in response to the growing threat of nuclear weapons proliferation. Further, in January 1994, President Clinton and Russia's President Yeltsin issued a Joint Statement Between the United States and Russia on Non-Proliferation of Weapons of Mass Destruction and the Means of Their Delivery. In accordance with these policies and statements, the focus of U.S. non-proliferation efforts is to ensure the safe, secure, long-term storage and disposition of surplus weapons-usable plutonium and highly enriched uranium (HEU). In July 1998, the United States and Russia signed a 5-year agreement to provide the scientific and technical basis for decisions concerning how surplus plutonium will be managed and a statement of principles with the intention of removing approximately 50 metric tons³ of plutonium from each country's stockpile. The Department is pursuing both the immobilization and mixed oxide (MOX) fuel approaches to surplus plutonium disposition, which include the siting, construction, operation, and deactivation of three facilities at one or two of four DOE candidate sites:

1. A facility for disassembling pits (a weapons component) and converting the recovered plutonium, as well as plutonium metal from other sources, into plutonium dioxide suitable for disposition. Candidate sites for this facility are the Hanford Site (Hanford) near Richland, Washington; Idaho National Engineering and Environmental Laboratory (INEEL) near Idaho Falls, Idaho; the Pantex Plant (Pantex) near Amarillo, Texas; and the Savannah River Site (SRS) near Aiken, South Carolina.

2. A facility for immobilizing surplus plutonium for eventual disposal in a geologic repository pursuant to the Nuclear Waste Policy Act. This facility would include a collocated capability for converting non-pit plutonium materials into plutonium dioxide suitable for immobilization. The immobilization facility would be located at either Hanford or SRS.

³ Some materials are already in a final disposition form (i.e., irradiated fuel) and will not require further action before disposal.

3. A MOX fuel fabrication facility for fabricating plutonium dioxide into MOX fuel. Candidate sites for this facility are Hanford, INEEL, Pantex, and SRS. Also part of the proposed action are MOX lead assembly⁴ activities at five candidate DOE sites: Argonne National Laboratory—West (ANL-W) at INEEL; Hanford; Lawrence Livermore National Laboratory (LLNL) in Livermore, California; Los Alamos National Laboratory (LANL) near Los Alamos, New Mexico; and SRS. The Department would fabricate a limited number of MOX fuel lead assemblies for testing in reactors before starting full production of MOX fuel under the proposed MOX fuel program. Post-irradiation examination activities would be performed at one of two sites, ANL-W or Oak Ridge National Laboratory (ORNL) in Oak Ridge, Tennessee.

In March 1999, DOE awarded a multi-phase contract to Duke Engineering & Services, COGEMA Inc., and Stone & Webster (collectively known as DCS) for the design, licensing, construction, operation, and eventual deactivation of the MOX fuel fabrication facility and for irradiating the MOX fuel. Full implementation of the base contract was contingent upon the successful completion of the National Environmental Policy Act (NEPA) process. The contract includes future provisions to use MOX fuel in six specific reactors: Catawba Nuclear Station Units 1 and 2 in South Carolina, McGuire Nuclear Station Units 1 and 2 in North Carolina, and North Anna Power Station Units 1 and 2 in Virginia.

DOE is aware that a decision to use surplus plutonium in MOX fuel could be perceived as a change in U.S. civilian fuel cycle policy. In fact, however, such a decision would not represent a change in policy. The United States does not encourage the civilian use of plutonium, and does not itself engage in reprocessing for the purposes of either nuclear explosives or nuclear power generation. Disposition of excess plutonium, regardless of the specific option chosen, will not change this basic fuel cycle policy.

NEPA Process

Surplus Plutonium Disposition Draft EIS

In December 1996, the Department published the Storage and Disposition PEIS. That PEIS analyzes the potential environmental consequences of alternative strategies for the long-term storage of weapons-usable plutonium and highly enriched uranium and the disposition of weapons-usable

⁴ A MOX lead assembly is a prototype reactor fuel assembly that contains MOX fuel.

plutonium that has been or may be declared surplus to national security needs.⁵ The Record of Decision (ROD) for the Storage and Disposition PEIS, issued on January 14, 1997, outlines DOE's decision to pursue an approach to plutonium disposition that would make surplus weapons-usable plutonium inaccessible and unattractive for weapons use. DOE's disposition strategy, consistent with the Preferred Alternative analyzed in the Storage and Disposition PEIS, allows for both the immobilization of some (and potentially all) of the surplus plutonium, and use of some of the surplus plutonium as MOX fuel in existing domestic, commercial reactors. The disposition of surplus plutonium would also involve disposal of both the immobilized plutonium and the MOX fuel (as spent nuclear fuel) in a potential geologic repository.⁶

On May 22, 1997, DOE published a Notice of Intent (NOI) in the Federal Register (FR) announcing its decision to prepare an EIS that would tier from the analysis and decisions reached in connection with the PEIS discussed above. The follow-on EIS, the Surplus Plutonium Disposition Environmental Impact Statement, addresses the extent to which each of the two plutonium disposition approaches (immobilization and MOX) would be implemented, and analyzes candidate sites for plutonium disposition facilities, as well as alternative technologies for immobilization.⁷ In July 1998, DOE issued the SPD Draft EIS. That draft included a description of the potential environmental impacts of using from three to eight commercial nuclear reactors to irradiate MOX fuel. The potential impacts were based on a generic reactor analysis presented in the Storage and Disposition PEIS. In March 1999, DOE awarded a contract, contingent on completion of the NEPA process, for MOX fuel fabrication and irradiation services, that identified the specific reactors that would be used to irradiate the MOX fuel. After this

contract award, DOE issued a Supplement to the SPD Draft EIS (Supplement) (April 1999) that describes the potential environmental impacts of using MOX fuel at the three proposed reactor sites. These site-specific analyses have been incorporated into the SPD Final EIS.

Alternatives Considered

The SPD EIS analyzes the potential environmental impacts associated with implementing pit disassembly and conversion of the recovered plutonium and clean plutonium metal at four candidate sites; conversion and immobilization of plutonium from non-pit sources at two candidate sites, and MOX fuel fabrication activities at four candidate sites. The SPD EIS also evaluates immobilizing plutonium in ceramic or glass forms, and compares the can-in-canister approach with the homogenous ceramic immobilization and vitrification approaches that were evaluated in the Storage and Disposition PEIS. As part of the MOX option, the SPD EIS also evaluates the potential impacts of fabricating MOX fuel lead assemblies (for test irradiation in domestic, commercial nuclear power reactors) at five candidate DOE sites, the impacts of subsequent post-irradiation examination of the lead assemblies at two candidate DOE sites, and the impacts of irradiating MOX fuel in domestic, commercial reactors.

Fifteen surplus plutonium disposition alternatives and the No Action Alternative are evaluated in the SPD EIS. These action alternatives are organized into 11 sets of alternatives, reflecting various combinations of facilities and candidate sites, as well as the use of new or existing buildings.

Each of the 15 alternatives includes a pit conversion facility, but the need for additional facilities in each alternative varies depending on the amount of plutonium to be immobilized. Eleven alternatives involve the hybrid approach of immobilizing 17 metric tons of surplus plutonium and using 33 metric tons for MOX fuel, and therefore require all three facilities. Four alternatives involve immobilizing all 50 metric tons, and therefore include only a pit conversion facility and an immobilization facility. The No Action Alternative does not involve disposition of surplus weapons-usable plutonium, but instead addresses continued storage of the plutonium in accordance with the Storage and Disposition PEIS Record of Decision (ROD), with the exception that DOE is now considering leaving the repackaged surplus pits in Zone 4 at Pantex for long-term storage in lieu of Zone 12 as originally planned.

Immobilization Technology Alternatives

The Storage and Disposition PEIS discusses several immobilization technologies, including the homogenous ceramic and vitrification alternatives that were evaluated in detail, as well as variants of those alternatives, which include the ceramic and glass can-in-canister approaches and a homogenous approach using an adjunct melter. The ROD for the Storage and Disposition PEIS states that DOE would make a determination on the specific technology on the basis of "the follow-on EIS." The SPD EIS is that follow-on EIS, and it identifies the ceramic can-in-canister approach as the preferred immobilization technology.

In order to bound the estimate of potential environmental impacts associated with ceramic and glass immobilization technologies, the Storage and Disposition PEIS analyzes the construction and operation of vitrification and ceramic immobilization facilities that employ a homogenous approach. These facilities are based on generic designs that do not involve the use of existing facilities or specific site locations. These generic designs allow for surplus plutonium to be immobilized in a homogenous form, either within a ceramic matrix and formed into disks, or vitrified as borosilicate glass logs.

In order to support a decision on the immobilization technology and form, the SPD EIS evaluates the potential environmental impacts of the ceramic and glass can-in-canister technologies, and compares those impacts with the impacts of the homogenous facilities evaluated in the Storage and Disposition PEIS. Hanford and SRS are the candidate sites for immobilization based on their existing plans for a high-level waste vitrification facility.

MOX Fuel Fabrication Alternatives

Alternatives that involve the fabrication of MOX fuel include the use of the fuel in existing domestic, commercial nuclear power reactors. The environmental impacts of using MOX fuel in these reactors are evaluated generically in the Storage and Disposition PEIS. When the SPD Draft EIS was published, the specific reactors were not known; therefore, the generic analysis from the Storage and Disposition PEIS was incorporated by reference in the SPD Draft EIS.

In May 1998, DOE initiated a procurement process to obtain MOX fuel fabrication and irradiation services. In compliance with its NEPA regulations in 10 CFR 1021.216, DOE requested that each offeror provide, as

⁵ DOE addressed the disposition of surplus highly enriched uranium in a separate environmental impact statement, the Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement, issued in June 1996, with the Record of Decision issued in July 1996.

⁶ The Nuclear Regulatory Commission has reviewed DOE's plans to phase immobilized material into the potential geologic repository, and has agreed that with adequate canister and package design features, the immobilized plutonium waste forms can be made acceptable for disposal in the repository.

⁷ The SPD EIS also analyzes a No Action Alternative, i.e., the possibility of disposition not occurring but, instead, continuing to store surplus plutonium in accordance with the Storage and Disposition PEIS ROD.

part of its proposal, environmental information specific to its proposed MOX facility design and the domestic, commercial reactors proposed to be used for irradiation of the fuel. That information was analyzed by the Department to identify potential environmental impacts of the proposals, and DOE's analysis was documented in an Environmental Critique prepared pursuant to 10 CFR 1021.216(g). That analysis was considered by the selection official as part of the award decision. DOE awarded a contract (contingent on completion of the NEPA process) to the team of Duke Engineering & Services, COGEMA Inc., and Stone & Webster (DCS) in March 1999 to provide the requested services. These services include design, licensing, construction, operation, and eventual deactivation of the MOX fuel fabrication facility, as well as irradiation of the MOX fuel in six domestic, commercial reactors. The reactors proposed by DCS are Duke Power Company's Catawba Nuclear Station, Units 1 and 2; and McGuire Nuclear Station, Units 1 and 2; and Virginia Power Company's North Anna Power Station, Units 1 and 2. Under the contract, no construction, fabrication, or irradiation of MOX fuel is authorized until the SPD EIS ROD is issued. Such site-specific activities, and DOE's exercise of contract options to allow those activities, would be contingent on decisions in this ROD.

Because the Environmental Critique contains proprietary information, it was not made available to the public. However, as provided in 10 CFR 1021.216(h), an Environmental Synopsis of the Environmental Critique was provided to the U.S. Environmental Protection Agency, made available to the public, and incorporated into the SPD EIS. Sections of the SPD EIS were revised or added to include reactor-specific information and were issued as a Supplement to the SPD Draft EIS. A Notice of Availability was published in the Federal Register on May 14, 1999 (64 FR 264019), providing a 45-day public comment period on the Supplement.⁸ This Supplement was distributed to the local reactor communities, to stakeholders who received the SPD Draft EIS, and others as requested.

Under the hybrid alternatives, DOE could produce up to 10 MOX fuel assemblies for testing in domestic, commercial reactors before commencement of full-scale MOX fuel fabrication, although it is likely that

only two lead assemblies would be needed.⁹ These lead assemblies would be available for irradiation to support NRC licensing and fuel qualification efforts. Potential impacts of MOX fuel lead assembly fabrication are analyzed for three of the candidate sites for MOX fuel fabrication (Hanford, ANL—W at INEEL, and SRS), and two additional sites, LANL and LLNL. Pantex was not considered for lead assembly fabrication because it does not currently have any facilities capable of MOX fuel fabrication. Post-irradiation examination of the lead assemblies would be conducted, if required, to support NRC licensing activities. Two potential sites for this activity are analyzed in the SPD EIS: ANL—W and Oak Ridge National Laboratory (ORNL). As discussed previously, DOE's preferred locations for lead assembly fabrication and post-irradiation examination are LANL and ORNL, respectively.

The Department also considered a No Action Alternative, as required by NEPA. In the No Action Alternative, surplus weapons-usable plutonium in storage at various DOE sites would remain at those locations. The vast majority of pits would continue to be stored at Pantex, and the remaining plutonium in various forms would continue to be stored at Hanford, INEEL, LLNL, LANL, Rocky Flats Environmental Technology Site (RFETS), and SRS.

Materials Analyzed

There are eight general categories used to describe the 50 metric tons of surplus plutonium analyzed in the SPD EIS, which represent the physical and chemical nature of the plutonium. Two of the categories—clean metal (including pits) and clean oxide—could either be fabricated into MOX fuel or immobilized. The remaining six categories of material—impure metals, plutonium alloys, impure oxides, uranium/plutonium oxides, alloy reactor fuel, and oxide reactor fuel—would be immobilized.

Preferred Alternative

As previously noted, DOE's Preferred Alternative for the disposition of surplus weapons-usable plutonium is analyzed as Alternative 3 in the SPD Final EIS. The Preferred Alternative encompasses the following:

⁹ The potential impacts of fabricating 10 lead assemblies and irradiating 8 of them were analyzed in the SPD EIS. Should fewer lead assemblies than analyzed be fabricated or irradiated, the potential impacts would be less than those described in the SPD EIS.

Pit Disassembly and Conversion at SRS (new construction)

Construct and operate a new pit conversion facility at SRS to disassemble nuclear weapons pits and convert the plutonium metal to a declassified oxide form suitable for international inspection and disposition using either the immobilization or the MOX/reactor approach. SRS is preferred for the pit conversion facility because the site has extensive experience with plutonium processing, and the pit conversion facility would complement existing missions and take advantage of existing infrastructure.

Immobilization at SRS (new construction and the Defense Waste Processing Facility)¹⁰

Construct and operate a new immobilization facility at SRS using the ceramic can-in-canister technology. This technology would immobilize plutonium in a ceramic form, seal it in cans, and place the cans in canisters filled with borosilicate glass containing intensely radioactive high-level waste at the existing Defense Waste Processing Facility (DWPF). This preferred can-in-canister approach at SRS would complement existing missions, take advantage of existing infrastructure and staff expertise, and enable DOE to use an existing facility (*i.e.*, DWPF).

Implementation of the can-in-canister approach would require the availability of sufficient quantities of high-activity radionuclides from SRS high-level waste to DWPF. Due to problems experienced with the In-Tank Precipitation process for separating high-activity radionuclides from liquid high-level waste, DWPF is currently operating with sludge feed, not liquid high-level waste. A thorough search for alternatives to the In-Tank Precipitation process has identified two viable processes (ion exchange and small tank precipitation) for separating the high-activity fraction from the liquid high-level waste and sending this fraction to DWPF. Extensive laboratory and bench scale testing has been conducted on both of these processes. Test results indicate that either process is capable of separating the high-activity radionuclides from the high-level waste and feeding those radionuclides to DWPF, although further research and development is necessary.¹¹ DOE is

¹⁰ The Savannah River Site was previously designated to be part of DOE's preferred alternative for immobilization in the Notice of Intent issued in May 1997.

¹¹ The National Research Council (the Council) is also evaluating a replacement technology for the In-Tank Precipitation process. The Council's study

Continued

⁸ On June 15, 1999, DOE held a public meeting in Washington, D.C., to receive comments on the Supplement to the SPD Draft EIS.

preparing a supplemental EIS on the proposed replacement of the In-Tank Precipitation process at SRS (NOI at 64 FR 8558, February 22, 1999).

Designation of a preferred process and construction of a pilot scale plant for scale-up of the preferred process are the next steps planned to resolve this issue.

In addition to these alternatives, the Department is analyzing the potential environmental impacts of another action alternative, direct grout, in light of technical and cost considerations. Under the direct grout alternative, the cesium component of the high-activity radionuclides would be entombed in grout rather than remain in the high-activity fraction provided to DWPF for vitrification and eventual disposal in a geologic repository. Therefore, the direct grout alternative would not provide the radiation barrier needed for surplus plutonium disposition using the can-in-canister technology at SRS. However, a DOE waste management requirement (DOE Manual 435.1, Radioactive Waste Management, Section II.B.2) provides that, for direct grout material to be disposed of as now being analyzed, "key radionuclides would have to be removed to the maximum extent that is technically and economically practical." This criterion would not be met in the event that any other action alternative is determined to be viable after further evaluation. Therefore, DOE regards the direct grout alternative as reasonable only if all of the other action alternatives analyzed in the supplemental EIS prove not to be viable.

In summary, although a specific method for providing the high-level waste needed for the can-in-canister immobilization alternatives for surplus plutonium disposition has not been determined, DOE is confident that an acceptable technical solution will be available at SRS. The ceramic can-in-canister approach would involve slightly lower environmental impacts than the homogenous approach. The ceramic can-in-canister approach would involve better performance in a potential geologic repository and provide greater proliferation resistance than the glass can-in-canister approach.

MOX Fuel Fabrication at SRS (new construction)

Construct and operate a new MOX facility at SRS and produce MOX fuel containing surplus weapons-usable

committee issued an interim report in October 1999. This committee recommends further research and development for the ion exchange and small tank precipitation alternatives, and for caustic side solvent extraction, a third process that would separate high-activity radionuclides that could be sent to DWPF.

plutonium for irradiation in existing domestic, commercial reactors. SRS is preferred for the MOX facility because this activity would complement existing missions and take advantage of existing infrastructure and staff expertise.

Lead Assembly Fabrication at LANL

Based on consideration of the capabilities of the candidate sites and input from the contractor team chosen for the MOX approach, DOE prefers LANL for lead assembly fabrication. LANL is preferred because it already has fuel fabrication facilities that would not require major modifications, and has existing site infrastructure and staff experience. Additionally, the surplus plutonium dioxide needed to fabricate the lead assemblies would already be on site (no transportation required).

Post-Irradiation Examination at ORNL

If post-irradiation examination is necessary for the purpose of qualifying the MOX fuel for commercial reactor use, DOE prefers to perform that task at ORNL. ORNL has the existing facilities and staff expertise needed to perform post-irradiation examination as a matter of its routine activities; no major modifications to facilities or processing capabilities would be required. In addition, ORNL is about 500 kilometers (km) from the reactor site that would irradiate the fuel (one of the reactors located at the McGuire Nuclear Station in North Carolina).

Environmental Impacts of Preferred Alternative

Chapter 4 and certain appendices of the SPD Final EIS analyze the potential environmental impacts of the surplus plutonium disposition alternatives in detail. The SPD Final EIS also evaluates the maximum impacts that would result at each of the potential disposition sites. Based on the analyses in the SPD Final EIS, including public comments on the SPD Draft EIS, the areas with impacts of most interest are as follows:

Disposition Facilities During Construction

Socioeconomics At its peak in 2003, construction of the three new surplus plutonium disposition facilities at SRS under this alternative would require 1,968 construction workers and should generate another 1,580 indirect jobs in the region. As the total employment increase of 3,548 direct and indirect jobs represents only 1.3 percent of the projected regional economic area (REA) workforce, it should have no major impact on the REA. Moreover, construction under the Preferred Alternative should have little impact on

the community services currently offered in the region of influence. In fact, it should help offset the 20 percent reduction in SRS's total workforce otherwise projected for the years 1997–2005.

Facility Accidents. The construction of new surplus plutonium disposition facilities at SRS could result in worker injuries or fatalities. DOE-required industrial safety programs would be in place to control the risks. Given the estimated 6,166 person-years of construction labor and standard industrial accident rates, approximately 610 cases of nonfatal occupational injury or illness and less than one fatality could be expected. As all construction would be in non-radiological areas, no radiological accidents should occur.

Cultural Resources. During conduct of the cultural resources impacts analysis for the Preferred Alternative, it was determined that construction of surplus plutonium disposition facilities at SRS could produce impacts on archaeological resources requiring mitigation. Archaeological investigations performed for the surplus plutonium disposition program discovered five archaeological sites in the proposed construction area. At least two of these sites have been recommended by DOE to the South Carolina State Historic Preservation Officer (SHPO) as eligible for nomination to the National Register of Historic Places. It appears that these sites were occupied during several different prehistoric periods, including the Late Woodland (A.D. 800–1000) and Mississippian (A.D. 1000–1600) Periods. These periods are poorly understood in the Central Savannah River Area. Therefore, these sites could contribute significantly to a better understanding of the Late Woodland and Mississippian Periods in this part of North America. Potential adverse impacts on these sites could be mitigated through either avoidance or data recovery. DOE currently plans to mitigate impacts by avoiding these sites.

Disposition Facilities During Operations

Socioeconomics. After construction, startup, and testing of the new SRS facilities in 2007, an estimated 1,120 new workers would be required to operate them. This level of employment should generate an additional 2,003 indirect jobs in the region. As the total employment requirement of 3,123 direct and indirect jobs represents 1 percent of the projected REA, it should have no major impact on the REA. Moreover, these jobs would have little impact on community services currently offered in

the region of influence. In fact, they should help offset the reduction in SRS's total workforce otherwise projected for the years 1997–2010 of 33 percent.

Facility Accidents (Impact to the public and workers). The most severe consequences of a design basis accident for the pit conversion facility would be associated with a tritium release; the most severe consequences for the immobilization and MOX facilities would be from a nuclear criticality. Bounding radiological consequences for the Maximally Exposed Individual (MEI)¹² are from the tritium release, which would result in a dose of 0.028 rem, corresponding to a latent cancer fatality (LCF) probability of 1.4×10^{-5} . A nuclear criticality of 10^{19} fissions would result in an MEI dose of 0.0016 rem from an accident at the immobilization facility and 0.016 rem from an accident at the MOX facility. Consequences of the tritium release accident for the general population in the environs of SRS would include an estimated 0.050 LCF. The frequency of either a tritium release or a criticality accident is estimated to be between 1 in 10,000 and 1 in 1,000,000 per year.

The combined radiological effects from total collapse of all three facilities in the beyond-design-basis earthquake would be approximately 18 LCFs. It should be emphasized that a seismic event of sufficient magnitude to collapse these facilities would likely cause the collapse of other DOE facilities, and would almost certainly cause widespread failure of homes, office buildings, and other structures in the surrounding area. The overall impact of such an event must therefore be seen in the context not only of the potential radiological impacts of these other facilities, but of hundreds, possibly thousands, of immediate fatalities from falling debris. The frequency of such an earthquake is estimated to be between 1 in 100,000 and 1 in 10,000,000 per year.

Surplus plutonium disposition operations at SRS could result in worker injuries and fatalities. DOE-required industrial safety programs would be in place to control the risks. Given the estimated employment of 11,535 person-years of labor and the standard DOE occupational accident rates, approximately 420 cases of nonfatal occupational injury or illness and 0.31 fatality could be expected for the duration of operations. If a criticality

occurred, workers within tens of meters could receive very high to fatal radiation exposures from the initial burst. The dose would strongly depend on the magnitude of the criticality, the distance from the criticality, and the amount of shielding provided by the structures and equipment between the workers and the accident.

Transportation. In all, approximately 2,500 shipments of radioactive materials would be carried out by DOE under the Preferred Alternative. The total distance traveled on public roads by trucks carrying radioactive materials would be 4.3 million kilometers.

The maximum foreseeable offsite transportation accident under this alternative (probability of occurrence: greater than 1 in 10 million per year) is a shipment of plutonium pits from one of DOE's storage locations to the pit conversion facility with a most severe (severity category VIII) accident in a rural population zone under neutral (average) weather conditions. If this accident were to occur, it could result in a dose of 87 person-rem to the public for an LCF risk of 0.044 and 96 rem to the hypothetical MEI for an LCF risk of 0.096. (The MEI, a hypothetical member of the general public, receives a larger dose than the public as a whole because it is unlikely that a person would be in position, and remain in position, to receive this hypothetical maximum dose.) No fatalities would be expected to occur. The probability of more severe accidents—e.g., less favorable weather conditions at the time of accident, or occurrence in a more densely populated area—was also evaluated, and estimated as lower than 1 chance in 10 million per year.

The total transportation accident risk was estimated by summing the risks (which takes account of both the probability and consequence of each type of accident) to the affected population from all hypothetical accidents. For the Preferred Alternative, that risk is as follows: a radiological dose to the population of 7 person-rem, resulting in a total population risk of 0.004 LCF; and traffic accidents resulting in 0.053 traffic fatality.

Irradiating MOX Fuel at Reactor Sites¹³

The environmental impacts described below are based on using a partial MOX

core (i.e., up to 40 percent MOX fuel) instead of a low enriched uranium (LEU) core at the Catawba Nuclear Station near York, South Carolina; the McGuire Nuclear Station near Huntersville, North Carolina; and the North Anna Power Station near Mineral, Virginia.

Reactor Accidents. There are differences in the expected risk of reactor accidents from the use of MOX fuel compared to the use of low enriched uranium fuel. The change in consequences to the surrounding population due to the use of MOX fuel is estimated to range from 9.0×10^{-4} fewer to 6.0×10^{-2} additional LCFs for design basis accidents, and from 7.0 fewer to 1,300 additional LCFs for beyond-design-basis accidents (16,900 versus 15,600 LCFs in the worst accident analyzed). Also, some of the beyond-design-basis accidents could result in prompt fatalities should they occur. The estimated increase in prompt fatalities due to MOX fuel being used during one of these accidents would range from no change to 28 additional fatalities (843 versus 815 prompt fatalities). As a result of these changes in projected consequences, there would be a change in the risk to the public associated with these accidents. The change in risk (in terms of an LCF or prompt fatality) to the surrounding population within 80 km (50 mi) of the proposed reactors is projected to range from a decrease of 6 percent to an increase of 3 percent in the risk of additional LCFs from design basis accidents, and from a decrease of 4 percent to an increase of 14 percent in the risk of additional prompt fatalities and LCFs from beyond-design-basis accidents.

The risk to the MEI would also change with the use of MOX fuel. Using MOX fuel during one of the design basis accidents evaluated is expected to change the MEI's chance of incurring an LCF from a decrease of 10 percent to an increase of 3 percent. The change in risk to the MEI of a prompt fatality or LCF as a result of using MOX fuel during one of the beyond-design-basis accidents evaluated is expected to range from a 1 percent increase to a 22 percent increase. In the most severe accident evaluated, an interfacing systems loss-of-coolant accident (ISLOCA), it is projected that the MEI would receive a fatal dose of radiation regardless of whether the reactor was using MOX fuel or LEU fuel at all of the proposed sites.

Beyond-design-basis accidents, if they were to occur, would be expected to result in major impacts to the reactors and the surrounding communities and environment, regardless of whether the

¹² The MEI is the hypothetical off-site person who has the highest exposure. This individual is assumed to be located at the point of maximum concentration of contaminants 24 hours a day, 7 days a week, for the period of operations under analysis.

¹³ The operators of the proposed reactors have indicated that little or no new construction would be needed to support the irradiation of MOX fuel at the sites. As a result, land use; visual, cultural, and paleontological resources; geology and soils; and site infrastructure would not be affected by any new construction or other activities related to MOX fuel use. Nor would there be any effect on air quality and noise, ecological and water resources, or socioeconomic.

reactor were using an LEU or partial MOX core. However, there is less than one chance in a million per year that a beyond-design-basis accident would actually happen, so the risk from these accidents is estimated to be low.

Lead Assembly and Post-Irradiation Examination Activities

The analysis of the potential impacts of conducting the lead assembly activities and post-irradiation examination indicates that little or no new construction or operational changes would be needed to support these activities. As a result, land use; visual, cultural, and paleontological resources; geology and soils; and site infrastructure would not be affected by any new construction or other activities related to lead assembly fabrication or post-irradiation examination. Nor would there be any effect on air quality and noise, ecological and water resources, or socioeconomics.

Avoidance and Minimization of Environmental Harm

For the Preferred Alternative, at SRS, storm water management and erosion control measures will be employed during construction of the disposition facilities. Cultural resources impacts will be mitigated either by avoidance or data recovery. Initial indications are the disposition facilities can be located in an area that will avoid disturbing known cultural resource areas.

During operation of the disposition facilities, radiation doses to individual workers will be kept at a minimum by maintaining comprehensive badged monitoring and "as low as reasonably achievable" (ALARA) programs during worker rotations. The storage facilities in the disposition buildings will be designed and operated in accordance with contemporary DOE orders and/or NRC regulations to reduce risks to workers and the public.

From a non-proliferation standpoint, the highest standards for safeguards and security will be employed during transportation, storage (*i.e.*, the stored weapons standard¹⁴) and disposition. DOE will coordinate the transport of surplus plutonium and fresh MOX fuel with State officials, consistent with contemporary policy. Although the actual routes will be classified, they will

be selected to circumvent populated areas where ever possible, maximize the use of interstate highways, and avoid bad weather. DOE will coordinate emergency preparedness plans and responses with involved states through liaison programs. The packaging, vehicles, and transport procedures being used are specifically designed and tested to prevent radiological release under all credible accident scenarios. The NRC regulates safeguards and security at facilities it licenses commensurate with the type of facility and type and amount of fissile or radioactive material present. Commercial nuclear power reactors have stringent regulations to prevent sabotage or diversion of special nuclear materials. Physical protection and safeguards and security will be ensured at the reactor sites by continued implementation of NRC requirements.

Environmentally Preferable Alternatives

The environmentally preferable alternative is the No Action Alternative. Under this alternative, surplus weapons-usable plutonium materials in storage at various DOE sites would remain at those locations. The vast majority of pits would continue to be stored at Pantex, and the remaining plutonium in various forms would continue to be stored at Hanford, INEEL, LLNL, LANL, RFETS, and SRS. The No Action Alternative would not satisfy the purpose and need for the proposed action because DOE's disposition decisions in the Storage and Disposition PEIS ROD would not be implemented. That ROD announced that, consistent with the Preferred Alternative in the Storage and Disposition PEIS, DOE had decided to reduce, over time, the number of locations where the various forms of plutonium are stored, through a combination of storage and disposition alternatives. Implementation of much of this decision requires the movement of surplus materials to disposition facility locations. Without disposition facilities, only pits that have been moved from RFETS to Pantex would be relocated in accordance with the Storage and Disposition PEIS ROD. All other surplus materials would continue to be stored indefinitely at their current locations, with the exception that DOE is considering leaving the repackaged surplus pits in Zone 4 at Pantex for long-term storage instead of zone 12 as originally planned. An appropriate environmental review will be conducted when the specific proposal for this change has been determined (*e.g.*, whether additional magazines need to be air-conditioned). The analysis in the

SPD EIS assumes that the surplus pits are stored in Zone 12 in accordance with the ROD for the Storage and Disposition PEIS.

Among the "action" alternatives analyzed in the SPD EIS, the environmentally preferable action alternative is the 50-Metric-Ton Immobilization Alternative with the Immobilization and Pit Conversion facilities located at SRS. This alternative would involve immobilizing all 50 metric tons of surplus plutonium at SRS. Under this alternative, only two facilities, the pit conversion facility and the immobilization facility, would be needed to accomplish the surplus plutonium disposition mission. Both the pit conversion and immobilization facilities would be new construction near the area currently designated for the Actinide Packaging and Storage Facility in F-Area. In addition, the canister receipt area at DWPF in S-Area would be modified to accommodate receipt and processing of the canisters transferred from the immobilization facility for filling with vitrified high-level waste. The pit conversion and immobilization facilities would be the same as those described for the Preferred Alternative, except that all the plutonium dioxide produced in the pit conversion facility would be transferred to the immobilization facility. To accommodate the additional 33 metric tons of plutonium that would be received from the pit conversion facility, the immobilization facility would be operated at a higher throughput (5 metric tons per year rather than 1.7 metric tons per year), and the operating workforce at the immobilization facility would be increased.

Comparison of Preferred Alternative to Other Alternatives

The Preferred Alternative requires the construction and operation of three new facilities; some minor modifications to, and work at, two existing DOE facilities; and use of existing domestic, commercial nuclear reactors for MOX fuel irradiation. The other hybrid alternatives would require the same facilities and activities; the immobilization-only alternatives would require the construction and operation of only two facilities. The environmentally preferable alternative, which is the No Action Alternative, does not involve construction or operation of any facilities, or use of new or existing facilities, other than those currently in use for the continued storage of the surplus plutonium. Furthermore, no transportation would be involved for the No Action

¹⁴The "Stored Weapons Standard" for weapons-usable fissile materials storage was initially defined in Management and Disposition of Excess Weapons Plutonium, National Academy of Sciences, 1994. DOE defines the Stored Weapons Standard as follows: The high standards of security and accounting for the storage of intact nuclear weapons should be maintained, to the extent practical, for weapons-usable fissile materials throughout dismantlement, storage, and disposition.

Alternative, and continued storage under this alternative would not affect any key environmental resource area at any of the seven storage locations. However, there would be doses to workers and the general population (and associated health effects) throughout the storage period at all of these locations. At SRS, the health effects from 50 years of storage under the No Action Alternative would be lower than those associated with implementation of the Preferred Alternative. Nonetheless, the Preferred Alternative would still contribute to the dose and associated health effects at locations where supporting activities like lead assembly fabrication and post-irradiation examination would occur.

The environmentally preferable action alternative, which is an immobilization-only alternative, would require the construction and operation of two, rather than three, facilities. For all of the key environmental resource areas except transportation and worker dose, the potential impacts of the Preferred Alternative are greater than for the environmentally preferable action alternative, although for most of the resource areas, the difference is less than 20 percent. The estimated LCFs and traffic fatalities are higher for the environmentally preferable action alternative, although both are well below one LCF. Worker dose is the same for both the preferred and the environmentally preferable action alternatives.

Relative ranking of the Preferred Alternative to other action alternatives varies by resource area. For all alternatives evaluated in the SPD EIS, the incremental concentrations of criteria air pollutant concentrations would be less than 2 percent of the applicable regulatory standard. The relative ranking of Preferred Alternative to the other action alternatives varies with the specific pollutant; for some, the Preferred Alternative ranks higher, for others, lower. The Preferred Alternative produces more, by approximately 5 to 25 percent, regulated waste than any of the other action alternatives.

All of the action alternatives would generate employment opportunities at each of the proposed facilities. In general, the Preferred Alternative requires the greatest number of construction and operation workers of all the action alternatives. However, for one alternative, approximately 5 percent more construction workers would be needed. The amount of land that would be disturbed for implementing any of the alternatives is relatively small. The Preferred Alternative requires the most land disturbance, and could potentially

affect cultural resource areas at SRS. However, as previously discussed in this ROD, DOE currently plans to mitigate impacts by avoiding sites that are eligible or potentially eligible for the National Register of Historic Places. SRS is the only candidate site at which cultural resource issues involving the proposed action have been identified. The action alternative with the least amount of land disturbance uses existing facilities at Hanford.

Because of the location of the proposed facilities relative to other activities at the sites, radiation doses would be received by construction workers at both INEEL and SRS. Doses to workers from construction and operation activities for each of the action alternatives could result in approximately 2.0 LCFs, with essentially no difference among any of the alternatives. There will be no dose (and therefore, no LCFs) to the general population for any of the action alternatives during construction of the proposed facilities. Although there is a small population dose associated with each of the action alternatives, no LCFs are expected to occur in the general population from routine operations for any of the alternatives. The most severe nonreactor design basis accident postulated for the Preferred Alternative, and all but one other action alternative, is a design basis fire in the pit conversion facility resulting in a tritium release. The resulting dose is highest for the Preferred Alternative, however, the associated dose would not be expected to result in any LCFs in the general population. None of the action alternatives is expected to result in traffic fatalities from nonradiological accidents or LCFs from radiological exposures or vehicle emissions. Impacts estimated for routine operations and postulated accidents at the reactor sites would be identical for all the hybrid alternatives.

Comments on Surplus Plutonium Disposition Final EIS

After issuing the SPD Final EIS, the Department received two letters. All of the issues raised in these letters have been covered in the body of the SPD Final EIS and in the Comment Response Document. The first letter contained a single comment requesting that the decision on a location for the lead assembly work retain the flexibility to allow doing the work at SRS. Based on consideration of the capabilities of the candidate sites and input from the team chosen for the MOX approach, the Department has decided to use LANL for fabrication of MOX fuel rods for use in fabrication of lead assemblies. LANL

was selected because it already has facilities that will not require major modifications for fuel rod fabrication, and takes advantage of existing infrastructure and staff experience. Additionally, the surplus plutonium dioxide needed to fabricate the MOX fuel rods for lead assemblies will already be on site.

The second letter contained numerous comments that opposed the use of MOX fuel in commercial power reactors. The commentor believes that the selection process of DCS and the commercial reactors was not opened to sufficient public scrutiny. The commentor repeated an earlier request that the Department hold additional public meetings in the vicinity of the three reactor sites before closing the public comment period, and that all information on the MOX project, including data submitted by DCS, DOE's Environmental Critique, and ORNL's data on expected radionuclide activities in MOX fuel, be made available to the public. During the public comment period on the Supplement to the SPD Draft EIS, which included specific reactor analyses, DOE held a public hearing in Washington, D.C., on June 15, 1999, and invited comments. While no additional hearings were held on the Supplement, other means were provided for the public to express their concerns and provide comments: mail; a toll-free telephone and fax line; and the Office of Fissile Materials Disposition Web-site. Also, at the invitation of South Carolina State Senator Phil Leventis, DOE attended and participated in a public hearing held on June 24, 1999, in Columbia, South Carolina.

Most of the information in DOE's Environmental Critique was included in the Environmental Synopsis released for public review; only proprietary and business-sensitive information was removed. The Duke, COGEMA, and Stone & Webster (DCS) team provided DOE with analyses of the environmental and computer modeling data, and population projections, but not the input data. The ratio of low-enriched uranium fuel to MOX fuel, provided by the Oak Ridge National Laboratory, is contained in the SPD Final EIS. Because the accident calculations are voluminous, they are not included in the SPD EIS. The calculations contain all of the input parameters including the MACCS2 computer files. Principal input parameters, such as accident source terms and population distributions, are included in the EIS.

The same commentor expressed concern that experience with the use of MOX fuel in the United States, as well

as internationally, is limited. The fabrication of MOX fuel and its use in commercial reactors has been accomplished in Western Europe. DOE would draw upon this experience in its disposition of the U.S. surplus plutonium. Electricite de France reactors in France have seen little or no impact from the use of MOX fuel on radionuclide releases in effluents. No change would be expected from normal operations, given that MOX fuel performs as well as LEU fuel and the fission products are retained within the fuel cladding. FRAGEMAs (a subsidiary of COGEMA and FRAMATOME) experience with fabricating MOX fuel indicates a fuel rod fission product leak rate of less than one-tenth of 1 percent. FRAGEMA has provided 1,253 MOX fuel assemblies, containing more than 300,000 fuel rods, for commercial reactor use. There have been no failures and leaks have occurred in only 3 assemblies (a total of 4 rods). All leaks occurred as a result of debris in the reactor coolant system and occurred in 1997 or earlier. French requirements for debris removal were changed in 1997 to alleviate these concerns. Since that time, there have been no leaks in MOX fuel rods. Further, as discussed in response DCR009-1 of the Comment Response Document, NRC would evaluate license applications and monitor the operations of the commercial reactors to ensure adequate margins of safety.

The commentor was also concerned that human and technical errors may lead to safety hazards at the reactors if MOX fuel is used. Particular safety issues were identified at McGuire, North Anna and Catawba (e.g., ice condenser problems and corrosion of service water pipes and auxiliary feedwater pipes). While the Department acknowledges that there are differences in the use of MOX fuel compared to LEU fuel, these differences are not expected to decrease the safety of the reactors. NRC has not considered it necessary to restrict operation of any of the other reactors in the United States that use ice condenser containments. All of the factors discussed by the commentor were evaluated by the proposed reactor licensees to ensure that the reactors, including those with ice condensers, can continue to operate safely using MOX fuel, and these factors will continue to be evaluated. Before any MOX fuel is used in the United States, NRC would have to perform a comprehensive safety review that would include information prepared by the reactor plant operators as part of their license amendment applications.

Another issue raised by the same commentor concerned the stability of plutonium compared to uranium and the alleged reduction in the ability to control the chain reaction when plutonium is added to the reactor in the form of MOX fuel. Differences between MOX fuel and uranium fuel are well characterized and can be accommodated through fuel and core design. All of the factors discussed by the commentor were evaluated by the proposed reactor licensees to ensure that the reactors can continue to operate safely using MOX fuel and will continue to be evaluated. Initial evaluations indicate that partial MOX fuel cores have a more negative fuel Doppler coefficient at hot zero power and hot full power, relative to LEU fuel cores for all times during the full cycle. These evaluations also indicate that partial MOX cores have a more negative moderator coefficient at hot zero power and hot full power, relative to LEU fuel cores for all times during the full cycle. These more negative temperature coefficients would act to shut the reactor down more rapidly during a heatup transient.

The commentor expressed concern that higher energy neutrons from plutonium are more likely to strike reactor parts such as the stainless steel containment vessel and degrade the metal parts of the reactor, resulting in embrittlement problems. Reactor vessel embrittlement is a condition in which the fast neutron fluence from the reactor core reduces the toughness (fracture resistance) of the reactor vessel metal. Analyses performed for the Department indicate that the core average fast flux in a partial MOX fuel core is comparable, within 3 percent, to the core average fast flux for a uranium fuel core. All of the reactors identified for the MOX mission have a comprehensive program of reactor vessel analysis and surveillance in place to ensure that NRC reactor vessel safety limits are not exceeded.

The commentor was also concerned that the use of MOX fuel would result in additional harmful radiation exposure to the public during a failure of the reactor containment structure. The commentor noted a study by the Nuclear Control Institute estimating that the risk to the public near McGuire or Catawba of contracting a deadly cancer following a severe accident will increase by nearly 40 percent when the plants start using plutonium fuel. DOE believes NCI's analysis overestimates the risk of using MOX fuel for two reasons. NCI's analysis did not account for the plutonium polishing step which has been added to the MOX fuel fabrication process. This step eliminates nearly all

of the americium from fresh MOX fuel, which significantly reduces the actinide inventory. In addition, NCI performed a generic reactor analysis while DOE performed plant specific analyses.

Analyses of a 40 percent weapons-grade MOX core indicate there would be approximately two times more americium-241 and plutonium-239, and slightly less than one and a half times the curium-242 than a reactor using LEU fuel. There are differences in the expected risk of reactor accidents from the use of MOX fuel. Some accidents would be expected to result in lower consequences to the surrounding population, and lower risks, while others would be expected to result in higher consequences and higher risks. There is an increase in risk, about 3 percent, for the large-break loss-of-coolant accident (the bounding design basis accident). The largest increase in risk for beyond-design-basis accidents is approximately 14 percent for an interfacing systems loss-of-coolant accident at North Anna. In the unlikely event that this beyond-design-basis accident were to occur, the expected number of LCFs would increase from 2,980 to 3,390 with a partial MOX core and prompt fatalities would increase from 54 to 60. Both of these accidents have an extremely low probability of occurrence. At North Anna, the likelihood of a large-break loss-of-coolant accident occurring is estimated at 1 chance in 48,000 per year and the likelihood of an interfacing systems loss-of-coolant accident occurring is estimated at 1 chance in 4.2 million per year.

Another issue raised by the commentor concerned timely and adequate emergency response to a MOX fuel accident due to limited resources of volunteer first responders. The subject of emergency response and subsequent cleanup of an accident that involves the release of nuclear materials is a topic of continuing discussion and planning between DOE and State, local, and tribal officials. Prior to any shipment of hazardous material, a transportation plan will be developed which includes details of emergency preparedness, security, and coordination of DOE with local emergency response authorities. Any additional training or equipment needed would be provided as part of the planning process. In addition, DOE maintains eight regional coordinating offices across the country, staffed 24 hours per day, 365 days per year to offer advice and assistance. Radiological Assistance Program teams are available to provide field monitoring, sampling, decontamination, communication, and other services.

As described in Appendix L of the SPD EIS, DOE anticipates that transportation required for the disposition of surplus plutonium would be done through DOE's Safe Secure Transport system. Since the establishment of the DOE Transportation Safeguards Division in 1975, the Safe Secure Transport system has transported DOE-owned cargo over more than 151 million kilometers (91 million miles) with no accidents causing a fatality or release of radioactive material.

Other Considerations

Cost Reports

To assist in the preparation of this ROD, DOE's Office of Fissile Materials Disposition prepared two cost reports. The first is Cost Analysis in Support of Site Selection for Surplus Weapons-Usable Plutonium Disposition (DOE/MD-0009; July 1998). This report provides site-specific cost information and analyses to support the selection of a preferred siting alternative for the alternatives considered in the SPD EIS. The second report is Plutonium Disposition Life Cycle Costs and Cost-Related Comment Resolution Document (DOE/MD-0013; November 1999). This report provides full life cycle costs for the Preferred Alternative as stated in the SPD EIS. It also contains the Department's responses to cost related comments submitted during the public review of the SPD Draft EIS.

Cost Analysis in Support of Site Selection

The summary costs listed below do not include the costs that would be the same, independent of where the facility is sited. Therefore, the costs are not full life cycle costs. The costs are presented in constant year 1997 dollars. Cost estimates for each of the required disposition facilities (Pit Disassembly and Conversion; MOX Fuel Fabrication; and Immobilization), including the additional supporting infrastructure, were created for each candidate site and were aggregated into two cost categories (1) design and construction and (2) operational. The cost estimates are considered to have an accuracy of plus or minus 40 percent for design, construction, and decommissioning, and an accuracy of plus or minus 20 percent for operations.

Hybrid Alternatives (Alternatives 2 through 10 in the SPD EIS). The estimated costs to design and construct the required facilities range from \$1.21 billion to \$1.40 billion, and estimated operational costs range from \$1.40 billion to \$1.58 billion. The total costs

for the hybrid alternatives range from \$2.67 billion to \$2.93 billion. The total cost of the hybrid alternatives would be reduced by the value of the MOX fuel provided to the participating reactors; at the time of this estimate the total cost after credit for the "fuel offset" was \$1.71 billion to \$2.01 billion.¹⁵

Immobilization-Only Alternatives (Alternatives 11 and 12 in the SPD EIS). The estimated costs to design and construct the required facilities range from \$0.73 billion to \$0.89 billion and the operational costs range from \$0.97 billion to \$1.0 billion. The Immobilization Only Alternatives range from \$1.71 billion to \$1.90 billion. The cost of the alternatives differ by approximately ten percent, well within the uncertainty of the cost estimates.

Life Cycle Cost for the Preferred Alternative

The summary cost listed below is the cost for the Preferred Alternative. The cost includes the cost of siting, construction, and operation of plutonium disposition facilities at DOE's Savannah River Site, as well as the cost associated with the irradiation of the MOX fuel in commercial reactors. In addition, the cost includes such costs as sunk (already spent) funds, and costs for developing and demonstrating the plutonium disposition technologies, transporting the plutonium and plutonium disposition products, start-up and deactivation and decommissioning of the three facilities. The costs are based upon the Cost Analysis in Support of Site Selection for Surplus Weapons-Usable Plutonium Disposition, DOE/MD-0009, July 22, 1998.

The total cost of implementing the Preferred Alternative is estimated to be \$4.07 billion in constant year 2000 dollars. The increase in cost over the 1998 estimate is primarily attributable to addition of life cycle costs specifically omitted from the 1998 cost report, technical program changes, specifically the increased size of the immobilization facility and the addition of the polishing step to the MOX fuel fabrication process, plus other cost changes (e.g., inflation).

Nonproliferation Assessment

To assist in the development of this ROD, DOE's Office of Arms Control and Nonproliferation, with support from the Office of Fissile Materials Disposition, prepared a report, Nonproliferation and

Arms Control Assessment of Weapons-Usable Fissile Material Storage and Plutonium Disposition Alternatives (DOE/NN-0007, January 1997). The report was issued in draft form in October 1996, and following a public comment period, was issued in final form in January 1997. It analyzes the nonproliferation and arms reduction implications of the alternatives for storage of plutonium and HEU, and disposition of excess plutonium. It is based in part on a Proliferation Vulnerability Red Team Report (SAND97-8203, UC-700, October 1996) prepared for the Office of Fissile Materials Disposition by Sandia National Laboratory. The assessment describes the benefits and risks associated with each option. Some of the "options" and "alternatives" discussed in the Nonproliferation Assessment are listed as "variants" (such as can-in-canister) in the Storage and Disposition Final PEIS. The following paragraphs discuss key conclusions of the report, as modified to meet current conditions.

Disposition of U.S. Excess Plutonium

Each of the alternatives for disposition of excess weapons plutonium that meets the Spent Fuel Standard¹⁶ would, if implemented appropriately, offer major nonproliferation and arms reduction benefits compared to leaving the material in storage in directly weapons-usable form. Taking into account the likely impact on Russian disposition activities, the no-action alternative appears to be by far the least desirable of the plutonium disposition options from a non-proliferation and arms reduction perspective.

Carrying out disposition of excess U.S. weapons plutonium, using alternatives that ensured effective non-proliferation controls and resulted in forms meeting the Spent Fuel Standard, would:

- Reduce the likelihood that current arms reductions would be reversed, by significantly increasing the difficulty, cost, and observability of returning this plutonium to weapons;
- Increase international confidence in the arms reduction process,

¹⁵ The MOX Fuel Fabrication Facility would produce nuclear fuel that will displace LEU fuel that utilities would otherwise purchase. The value of this fuel, deemed the MOX fuel offset, is estimated to be \$920 million.

¹⁶ "Spent Fuel Standard" is a term coined by the National Academy of Sciences (NAS, 1994, Management and Disposition of Excess Weapons Plutonium, National Academy Press, Washington, D.C., pg 12) and modified by DOE (glossary from Office of Fissile Materials Disposition web site at <http://www.doe-md.com>) denoting the main objective of alternatives for the disposition of surplus plutonium: that such plutonium be made roughly as inaccessible and unattractive for weapons use as the much larger and growing stock of plutonium in civilian spent fuel.

strengthening political support for the non-proliferation regime and providing a base for additional arms reductions, if desired;

- Reduce long-term proliferation risks posed by this material by further helping to ensure that weapons-usable material does not fall into the hands of rogue states or terrorist groups; and
- Lay the essential foundation for parallel disposition of excess Russian plutonium, reducing the risks that Russia might threaten U.S. security by rebuilding its Cold War nuclear weapons arsenal, or that this material might be stolen for use by potential proliferators.

Choosing the "no-action alternative" of leaving U.S. excess plutonium in storage in weapons-usable form indefinitely, rather than carrying out disposition:

- Would represent a clear reversal of the U.S. position seeking to reduce excess stockpiles of weapons-usable materials worldwide;
- Would make it impossible to achieve disposition of Russian excess plutonium;
- Could undermine international political support for non-proliferation efforts by leaving open the question of whether the United States was maintaining an option for rapid reversal of current arms reductions; and
- Could undermine progress in nuclear arms reductions.

The benefits of placing U.S. excess plutonium under international monitoring and then transforming it into forms that met the Spent Fuel Standard would be greatly increased, and the risks of these steps significantly decreased, if Russia took comparable steps with its own excess plutonium on a parallel track. The two countries need not use the same plutonium disposition technologies. However, as the 1994 NAS committee report concluded, options for disposition of U.S. excess weapons plutonium will provide maximum nonproliferation and arms control benefits if they:

- Minimize the time during which the excess plutonium is stored in forms readily usable for nuclear weapons;
- Preserve material safeguards and security during the disposition process, seeking to maintain to the extent possible the same high standards of security and accounting applied to stored nuclear weapons (the Stored Weapons Standard);
- Result in a form in which the plutonium would be as inaccessible and unattractive for weapons use as the larger and growing quantity of plutonium in commercial spent fuel (the Spent Fuel Standard).

In order to achieve the benefits of plutonium disposition as rapidly as possible, and to minimize the risks and negative signals resulting from leaving the excess plutonium in storage, it is important for disposition options to begin, and to complete the mission as soon as practicable, taking into account non-proliferation, environment, safety, and health, and economic constraints. Timing should be a key criterion in judging disposition alternatives. Beginning the disposition quickly is particularly important to establishing the credibility of the process, domestically and internationally.

Each of the alternatives under consideration for plutonium disposition:

- Has its own advantages and disadvantages with respect to non-proliferation and arms control, but none is clearly superior to the others;
- Can potentially provide high levels of security and safeguards for nuclear materials during the disposition process, mitigating the risk of theft of nuclear materials; and
- Can potentially provide for effective international monitoring of the disposition process.

Plutonium disposition can only reduce, not eliminate, the security risks posed by the existence of excess plutonium, and will involve some risks of its own. Because all plutonium disposition alternatives would take decades to complete, disposition is not a near-term solution to the problem of nuclear theft and smuggling. While disposition will make a long-term contribution, the near-term problem must be addressed through programs to improve security and safeguarding for nuclear materials, and to ensure adequate police, customs, and intelligence capabilities to interdict nuclear smuggling. All plutonium disposition alternatives under consideration would involve processing and transport of plutonium, which will involve more risk of theft in the short term than if the material had remained in heavily guarded storage, in return for the long-term benefit of converting the material to more proliferation-resistant forms.

Both the United States and Russia will still retain substantial stockpiles of nuclear weapons and weapons-usable fissile materials after disposition of the fissile materials currently considered excess is complete. These weapons and materials will continue to pose a security challenge regardless of what is done with excess plutonium. None of the disposition alternatives under consideration would make it impossible to recover the plutonium for use in

nuclear weapons, or make it impossible to use other plutonium to rebuild a nuclear arsenal. Therefore, disposition will only reduce, not eliminate, the risk of reversal of current nuclear arms reductions. A United States decision to choose reactor alternatives for plutonium disposition could offer additional arguments and justifications to those advocating plutonium reprocessing and recycle in other countries. This could increase the proliferation risk if it in fact led to significant additional separation and handling of weapons-usable plutonium. On the other hand, if appropriately implemented, plutonium disposition might also offer an opportunity to develop improved procedures and technologies for protecting and safeguarding plutonium, which could reduce proliferation risks and would strengthen United States efforts to reduce the stockpiles of separated plutonium in other countries.

Large-scale bulk processing of plutonium, including processes to convert plutonium pits to oxide and prepare other forms for disposition, as well as fuel fabrication or immobilization processes, represents the stage of the disposition process when material is most vulnerable to covert theft by insiders or covert diversion by the host state. However, such bulk processing is required for all disposition alternatives. In particular, initial processing of plutonium pits and other forms is among the most proliferation sensitive stages of the disposition process, but it is largely common to all the options.

Transport of plutonium is the point in the disposition process when the material is most vulnerable to overt armed attacks designed to steal plutonium. With sufficient resources devoted to security, however, high levels of protection against such overt attacks can be provided.

Conclusions Relating to Specific Disposition Technologies

Reactor technology will meet the Spent Fuel Standard. Reactor technology has some advantage over the immobilization technology with respect to perceived irreversibility, in that the plutonium would be converted from weapons-grade to reactor-grade, even though it is possible to produce nuclear weapons with both weapons and reactor-grade plutonium. However, the immobilization technology has some advantage over the reactor technology in avoiding the perception that the latter approach could potentially encourage additional separation and civilian use of plutonium, which itself poses

proliferation risks. Because reactor technology results in accountable "items" (for purposes of international safeguards) whose plutonium content can be accurately measured, this approach offers some advantage in accounting to ensure that the output plutonium matches the input plutonium from the process. The principal uncertainty with respect to using excess weapons plutonium as MOX fuel in domestic reactors relates to the potential difficulty of gaining political and regulatory approvals for the various operations required.

Immobilization technology (can-in-canister) is being refined resulting in an increase in the resistance to separation of the plutonium cans from the surrounding glass, with the goal of meeting the Spent Fuel Standard. The immobilization options have the potential to be implemented more quickly than the reactor options. They face somewhat less political uncertainty but somewhat more technical uncertainty than the reactor options.

The "can-in-canister" immobilization options have a timing advantage over the homogeneous immobilization options, in that, by potentially relying on existing facilities, they could begin several years sooner. As noted above, however, modified systems intended to allow this option to meet the Spent Fuel Standard are still being designed.

Decisions ¹⁷

Consistent with the January 1997 decision on the Storage and Disposition PEIS, the Department of Energy is affirming its decision to use a hybrid approach for the safe and secure disposition of up to 50 metric tons of surplus plutonium using both immobilization and mixed oxide fuel technologies and to construct and operate three new facilities at its Savannah River Site. The hybrid approach allows for the immobilization of approximately 17 metric tons of surplus plutonium and the use of up to 33 metric tons as mixed oxide fuel which would be irradiated in commercial reactors.

Construction and Operation of a Pit Disassembly and Conversion Facility

Consistent with the Preferred Alternative in the SPD Final EIS, the Department has decided to construct

and operate a new pit conversion facility at SRS for the purpose of disassembling nuclear weapons pits and converting the plutonium metal to a declassified oxide form suitable for international inspection and disposition, using either immobilization or MOX/reactor approaches. SRS was selected for the pit conversion facility because the site has extensive experience with plutonium processing, and the pit conversion facility complements existing missions and takes advantage of existing infrastructure.

Construction and Operation of an Immobilization Facility and Selection of an Immobilization Technology ¹⁸

Consistent with the Preferred Alternative in the SPD Final EIS, the Department has decided to construct and operate a new immobilization facility at SRS using the ceramic can-in-canister technology. This technology will be used to immobilize approximately 17 metric tons of surplus plutonium in a ceramic form, seal it in cans, and place the cans in canisters filled with borosilicate glass containing intensely radioactive high-level waste at the existing Defense Waste Processing Facility. The decision is based, in part, on the fact that the can-in-canister approach at SRS complements existing missions, takes advantage of existing infrastructure and staff expertise, and enables DOE to use an existing facility (DWPF). The ceramic can-in-canister approach will also provide better performance in a geologic repository and provide greater proliferation resistance than the glass can-in-canister approach.

Construction and Operation of a Mixed Oxide Fuel Fabrication Facility and Irradiation in Commercial Reactors

Consistent with the Preferred Alternative in the SPD Final EIS, the Department has decided to construct and operate a new facility at SRS to produce MOX fuel containing up to 33 metric tons of surplus weapons-usable plutonium for irradiation in existing domestic, commercial reactors. The decision to use SRS is made, in part, because this activity complements existing missions and takes advantage of existing infrastructure and staff expertise. Based on this selection, the

Department will authorize DCS to fully implement the base contract.

As previously stated in the Storage and Disposition PEIS ROD (62 FR 3014, January 21, 1997), the use of MOX fuel in existing reactors will be undertaken in a manner that is consistent with the United States' policy objective on the irreversibility of the nuclear disarmament process and the United States' policy discouraging the civilian use of plutonium. To this end, implementing the MOX alternative will include government ownership and control of the MOX fuel fabrication facility at a DOE site, and use of the facility only for the surplus plutonium disposition program. There will be no reprocessing or subsequent reuse of spent MOX fuel. The MOX fuel will be used in a once-through fuel cycle in existing reactors, with appropriate arrangements, including contractual or licensing provisions limiting use of MOX fuel to surplus plutonium disposition.

Selection of a Site for Lead Assembly Fabrication

Consistent with the Preferred Alternative in the SPD EIS, the Department has decided to use LANL for fabrication of MOX fuel rods for use in fabrication of lead assemblies. Based on consideration of the capabilities of the candidate sites and input from the team chosen for the MOX approach, LANL was selected because it already has facilities (*i.e.*, Technical Area 55) that will not require major modifications in order to fabricate fuel rods, and takes advantage of existing infrastructure and staff experience. Additionally, the surplus plutonium dioxide needed to fabricate the MOX fuel rods for lead assemblies will already be on site.

At this time, however, no decision is being made as to which facility at LANL will be used for final assembly of the MOX fuel rods into lead assemblies. DOE is currently evaluating whether there may be the need for additional environmental analysis to support the final stages of lead assembly fabrication at LANL. Pending completion of that review, DOE is deferring a decision as to where on the LANL site this final lead assembly work will be done.

Selection of a Site for Post-Irradiation Examination of Lead Assemblies

If post-irradiation examination is necessary for the purpose of qualifying the MOX fuel for commercial reactor use, the Department has decided to perform that task at ORNL, consistent with the Preferred Alternative in the SPD Final EIS. ORNL has the existing

¹⁷ Included in these decisions is the Department's decision to fulfill the Moscow Nuclear Safety and Security agreement to apply International Atomic Energy Agency safeguards to surplus plutonium as soon as it is practical. Further, consistent with a Presidential Directive, the Department is continuing to work towards maximizing the quantities of materials eligible for International Atomic Energy Agency safeguards.

¹⁸ The Department intends to use essentially all of the plutonium oxide produced by the Pit Disassembly and Conversion Facility as feed material for mixed oxide fuel. However, some small amounts may be unsuitable for this purpose and will be shipped to the Immobilization Facility for disposition.

facilities and staff expertise needed to perform post-irradiation examination as a matter of its routine activities and no major modifications to facilities or processing capabilities would be required. In addition, ORNL is only about 500 km from the reactor site that would irradiate the fuel, considerably closer than ANL—W, which is about 3,700 km away.

Use of MOX Fuel in Canadian Uranium Deuterium Reactors

In the Storage and Disposition PEIS ROD, DOE retained the option to use some of the surplus plutonium as MOX fuel in Canadian Uranium Deuterium (CANDU) reactors, which would have been undertaken only in the event that a multilateral agreement were negotiated among Russia, Canada, and the United States. Since the SPD Draft EIS was issued, DOE determined that adequate reactor capacity is available in the United States for disposition of that portion of the U.S. surplus plutonium suitable for MOX fuel. Therefore, DOE is no longer actively pursuing the CANDU option. However, the CANDU option is still being considered for the disposition of Russian surplus plutonium. To assist U.S., Russia, and Canada in considering this option the three countries are jointly conducting an experiment which will involve irradiating MOX fuel pins that have been fabricated from U.S. and Russian surplus weapons plutonium in a Canadian research reactor. This effort involves a one-time shipment of a small quantity of weapons plutonium from the U.S. to Canada.

Conclusion

The Department of Energy has decided to disposition up to 50 metric tons of plutonium at SRS using a hybrid approach that involves both the ceramic can-in-canister immobilization approach and the MOX fuel approach. Approximately 17 metric tons of surplus plutonium will be immobilized in a ceramic form, placed in cans, and embedded in large canisters containing high-level vitrified waste for ultimate disposal in a geologic repository pursuant to the Nuclear Waste Policy Act. Approximately 33 metric tons of surplus plutonium will be used to fabricate MOX fuel, which will be irradiated in existing domestic, commercial reactors. The reactors are the Catawba Nuclear Station near York, South Carolina; the McGuire Nuclear Station near Huntersville, North Carolina; and the North Anna Power Station near Mineral, Virginia. The resulting spent fuel will be placed in a geologic repository pursuant to the

Nuclear Waste Policy Act. Pursuing this hybrid approach provides the best opportunity for U.S. leadership in working with Russia to implement similar options for reducing Russia's excess plutonium in parallel. Further, it sends the strongest possible signal to the world of U.S. determination to reduce stockpiles of surplus weapons-usable plutonium as quickly as possible and in an irreversible manner. Pursuing both immobilization and MOX fuel fabrication also provides important insurance against uncertainties of implementing either approach by itself. The construction of new facilities for the disposition of surplus U.S. plutonium would not take place unless there is significant progress on plans for plutonium disposition in Russia. In the plutonium disposition effort, the United States will work with Russia to develop acceptable methods and technologies for transparency measures, including appropriate international verification measures and stringent standards of physical protection, control, and accounting for the management of surplus plutonium.

Issued in Washington, DC, January 4, 2000.

Bill Richardson,

Secretary.

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DEPARTMENT OF ENERGY

Docket Nos. FE C&E 99-27, C&E 99-28, C&E 99-29, C&E 99-30 & C&E 99-31

Office of Fossil Energy; Notice of Filings of Coal Capability of Cleco Evangeline LLC, Liberty Electric Power, LLC, ANP Bellingham Energy Co., Midlothian Energy Limited Partnership and La Paloma Generating Company, LLC; Powerplant and Industrial Fuel Use Act

AGENCY: Office of Fossil Energy, Department of Energy.

ACTION: Notice of filings.

SUMMARY: Cleco Evangeline LLC, Liberty Electric Power, LLC, ANP Bellingham Energy Company, Midlothian Energy Limited Partnership and La Paloma Generating Company, LLC have submitted coal capability self-certifications pursuant to section 201 of the Powerplant and Industrial Fuel Use Act of 1978, as amended.

ADDRESSES: Copies of self-certification filings are available for public inspection, upon request, in the Office of Coal & Power Im/Ex, Fossil Energy, Room 4G-039, FE-27, Forrestal

Building, 1000 Independence Avenue, SW, Washington, DC 20585.

FOR FURTHER INFORMATION CONTACT: Ellen Russell at (202) 586-9624

SUPPLEMENTARY INFORMATION: Title II of the Powerplant and Industrial Fuel Use Act of 1978 (FUA), as amended (42 U.S.C. 8301 *et seq.*), provides that no new baseload electric powerplant may be constructed or operated without the capability to use coal or another alternate fuel as a primary energy source.

In order to meet the requirement of coal capability, the owner or operator of such facilities proposing to use natural gas or petroleum as its primary energy source shall certify, pursuant to FUA section 201(d), to the Secretary of Energy prior to construction, or prior to operation as a base load powerplant, that such powerplant has the capability to use coal or another alternate fuel. Such certification establishes compliance with section 201(a) as of the date filed with the Department of Energy. The Secretary is required to publish a notice in the Federal Register that a certification has been filed. The following owners/operators of proposed new baseload powerplants have filed a self-certification in accordance with section 201(d).

Owner: Cleco Evangeline LLC (C&E 99-27).

Operator: Cleco Evangeline LLC.

Location: Evangeline Parish, Louisiana.

Plant Configuration: Combined-cycle.

Capacity: 710 MW.

Fuel: Natural gas.

Purchasing Entities: Williams Energy Marketing & Trading Co.

In-Service Date: June 1, 2000.

Owner: Liberty Electric Power, LLC (C&E 99-28).

Operator: Liberty Electric Power, LLC.

Location: Delaware County, PA.

Plant Configuration: Combined-cycle.

Capacity: 500 MW.

Fuel: Natural gas.

Purchasing Entities: To be determined.

In-Service Date: Fourth quarter, 2001.

Owner: ANP Bellingham Energy Company (C&E 99-29).

Operator: ANP Bellingham Energy Company.

Location: Bellingham, MA.

Plant Configuration: Combined-cycle.

Capacity: 570 MW.

Fuel: Natural gas.

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